Department of Energy
Washington, DC 20585

July 14, 2000

Dear Interested Party:

The *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE/EIS-0306) is enclosed for your information. This document has been prepared in accordance with the National Environmental Policy Act (NEPA) and reflects comments received on a draft released in July 1999.

The Department of Energy (DOE) proposes to treat its inventory of sodium-bonded spent nuclear fuel and facilitate its eventual disposition in a future geologic repository. The environmental impact statement (EIS) evaluates the associated potential environmental impacts at one or more spent nuclear fuel management facilities. The EIS analyzes the melt and dilute, electrometallurgical, plutonium-uranium extraction treatment technologies, and packaging in high-integrity cans as treatment alternatives as well as a no-action alternative.

After careful consideration of public comments and programmatic, environmental, nonproliferation, and cost issues, DOE has selected electrometallurgical treatment as its preferred alternative for the treatment and management of all sodium-bonded spent nuclear fuel except for the Fermi-1 blanket fuel. The physical characteristics of the Fermi-1 blanket spent nuclear fuel are such that alternative treatment techniques that currently require additional development may be more appropriate to treat this particular spent fuel. DOE will investigate those alternative techniques and make a final decision regarding the Fermi-1 blanket fuel at a later date.

The final EIS is available on the Office of Nuclear Energy, Science and Technology Web site (www.ne.doe.gov), DOE's NEPA Web site (tis.eh.doe.gov/NEPA), at libraries at the University of South Carolina and University of New Mexico, and at DOE reading rooms in Idaho Falls, Idaho; Aiken, South Carolina; Oak Ridge, Tennessee; Richland, Washington; and Washington, D.C.

Additional copies of the final EIS and the National Research Council's *Electrometallurgical Techniques for DOE Spent Fuel Treatment, Final Report (April 2000)* are available upon request by calling 1-877-450-6904 or by sending an e-mail to sodium.fuel.eis@hq.doe.gov.

We appreciate your continued participation in this decision-making process.

Sincerely,

William D. Magwood, IV, Director
Office of Nuclear Energy, Science
and Technology

Enclosure

Printed with soy ink on recycled paper

DOE/EIS - 0306
July 2000

Final Environmental Impact Statement

for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Volume 1

U.S. Department of Energy
Office of Nuclear Energy,
Science and Technology
Washington, DC 20585

In response to comments on the SBSNF Draft EIS and as a result of information that was unavailable at the time of the issuance of the draft EIS, the final EIS contains revisions and new information. These revisions and new information are indicated by a double underline for minor word changes or by a sidebar in the margin for sentence or larger additions. Appendix A contains the comments received during the public review period of the SBSNF Draft EIS and DOE's responses to these comments.

AVAILABILITY OF THE DRAFT SBSNF EIS

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COVER SHEET

Responsible Agency: United States Department of Energy (DOE)

Title: Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel (SBSNF EIS)

Locations: Idaho, South Carolina

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***Abstract*:** DOE is responsible for the safe and efficient management of its sodium-bonded spent nuclear fuel. This fuel contains metallic sodium, a highly reactive material; metallic uranium, which is also reactive; and in some cases, highly enriched uranium. The presence of reactive materials could complicate the process of qualifying and licensing DOE's sodium-bonded spent nuclear fuel inventory for disposal in a geologic repository. Currently, more than 98 percent of this inventory is located at the Idaho National Engineering and Environmental Laboratory (INEEL), near Idaho Falls, Idaho. In addition, in a 1995 agreement with the State of Idaho, DOE committed to remove all spent nuclear fuel from Idaho by 2035. This EIS evaluates the potential environmental impacts associated with the treatment and management of sodium-bonded spent nuclear fuel in one or more facilities located at Argonne National Laboratory-West (ANL-W) at INEEL and either the F-Canyon or Building 105-L at the Savannah River Site (SRS) near Aiken, South Carolina. DOE has identified and assessed six proposed action alternatives in this EIS. These are: (1) electrometallurgical treatment of all fuel at ANL-W, (2) direct disposal of blanket fuel in high-integrity cans with the sodium removed at ANL-W, (3) plutonium-uranium extraction (PUREX) processing of blanket fuel at SRS, (4) melt and dilute processing of blanket fuel at ANL-W, (5) melt and dilute processing of blanket fuel at SRS, and (6) melt and dilute processing of all fuel at ANL-W. In addition, Alternatives 2 through 5 include the electrometallurgical treatment of driver fuel at ANL-W. Under the No Action Alternative, the EIS evaluates both the continued storage of sodium-bonded spent nuclear fuel until the development of a new treatment technology or direct disposal without treatment. Under all of the alternatives, the affected environment is primarily within 80 kilometers (50 miles) of spent nuclear fuel treatment facilities. Analyses indicate little difference in the environmental impacts among alternatives. DOE has identified electrometallurgical treatment as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel.

***Public Comments*:** The draft EIS was issued for public review and comment on July 31, 1999. The comment period ended on September 28, 1999, although late comments were accepted. Public hearings to solicit comments on the draft EIS were held in North Augusta, South Carolina; Boise and Idaho Falls, Idaho; and Arlington, Virginia. All comments were considered during the preparation of the final EIS, which also incorporates additional and new information received since the issuance of the draft EIS. In response to comments on the SBSNF Draft EIS and as a result of information that was unavailable at the time of the issuance of the draft EIS, the final EIS contains revisions and new information. These revisions and new information are indicated by a double underline for minor word changes or by a sidebar in the margin for sentence or larger additions. Appendix A contains the comments received during the public review period of the SBSNF Draft EIS and DOE's responses to these comments. DOE will use the analyses presented in this final EIS as well as other information in preparing the Record of Decision for the treatment and management of its sodium-bonded spent nuclear fuel. DOE will issue this Record of Decision no sooner than 30 days after the U.S. Environmental Protection Agency publishes a notice of availability of this final EIS in the *Federal Register*.

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ACRONYMS, ABBREVIATIONS, AND CONVERSION CHARTS

ANL-W	Argonne National Laboratory-West
BEIR	Biological Effects of Ionizing Radiation
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
CPP	Chemical Processing Plant
DOE	U.S. Department of Energy
EBR-II	Experimental Breeder Reactor-II
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
ERPG	Emergency Response Planning Guideline
FR	*Federal Register*
GMODS	Glass Material Oxidation and Dissolution System
HFEF	Hot Fuel Examination Facility
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
MEDEC	melt, drain, evaporate, and calcine (ANL-W process)
NAAQS	National Ambient Air Quality Standards
NEPA	National Environmental Policy Act
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
OSHA	Occupational Safety and Health Administration
PM_n	Particulate matter less than or equal to *n* microns in diameter
PUREX	Plutonium-Uranium Extraction
RCRA	Resource Conservation and Recovery Act
SBSNF	Sodium-Bonded Spent Nuclear Fuel
SRS	Savannah River Site
U.S.C.	United States Code
YAG	yttrium-aluminum-garnet

Metric Conversion Chart

To Convert Into Metric			*To Convert From Metric*		
If You Know	**Multiply By**	**To Get**	**If You Know**	**Multiply By**	**To Get**
Length					
inches	2.54	centimeters	centimeters	0.3937	inches
feet	30.48	centimeters	centimeters	0.0328	feet
feet	0.3048	meters	meters	3.281	feet
yards	0.9144	meters	meters	1.0936	yards
miles	1.60934	kilometers	kilometers	0.6214	miles
Area					
square inches	6.4516	square centimeters	square centimeters	0.155	square inches
square feet	0.092903	square meters	square meters	10.7639	square feet
square yards	0.8361	square meters	square meters	1.196	square yards
acres	0.40469	hectares	hectares	2.471	acres
square miles	2.58999	square kilometers	square kilometers	0.3861	square miles
Volume					
fluid ounces	29.574	milliliters	milliliters	0.0338	fluid ounces
gallons	3.7854	liters	liters	0.26417	gallons
cubic feet	0.028317	cubic meters	cubic meters	35.315	cubic feet
cubic yards	0.76455	cubic meters	cubic meters	1.308	cubic yards
Weight					
ounces	28.3495	grams	grams	0.03527	ounces
pounds	0.4536	kilograms	kilograms	2.2046	pounds
short tons	0.90718	metric tons	metric tons	1.1023	short tons
Temperature					
Fahrenheit	Subtract 32, then multiply by 0.55556	Celsius	Celsius	Multiply by 1.8, then add 32	Fahrenheit

Metric Prefixes

Prefix	*Symbol*	*Multiplication Factor*
exa-	E	1 000 000 000 000 000 000 = 10^{18}
peta-	P	1 000 000 000 000 000 = 10^{15}
tera-	T	1 000 000 000 000 = 10^{12}
giga-	G	1 000 000 000 = 10^{9}
mega-	M	1 000 000 = 10^{6}
kilo-	k	1 000 = 10^{3}
hecto-	h	100 = 10^{2}
deka-	da	10 = 10^{1}
deci-	d	0.1 = 10^{-1}
centi-	c	0.01 = 10^{-2}
milli-	m	0.001 = 10^{-3}
micro-	µ	0.000 001 = 10^{-6}
nano-	n	0.000 000 001 = 10^{-9}
pico-	p	0.000 000 000 001 = 10^{-12}
femto-	f	0.000 000 000 000 001 = 10^{-15}
atto-	a	0.000 000 000 000 000 001 = 10^{-18}

1. INTRODUCTION

Chapter 1 provides an overview of the U.S. Department of Energy's proposal for treatment and management of sodium-bonded spent nuclear fuel. This chapter discusses the background, purpose and need for agency action, and scope of the *Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*. Included are discussions on the decisions to be made and issues identified by the public during the scoping and public comment periods. The chapter concludes with sections on the relationship of this proposal to other actions and programs under the National Environmental Policy Act and the organization of the document.

1.1 BACKGROUND

For nearly four decades, research, development, and demonstration activities associated with liquid metal fast breeder reactors were conducted at the Experimental Breeder Reactor-II (EBR-II), about 40 miles west of Idaho Falls, Idaho; the Enrico Fermi Atomic Power Plant[1] in Monroe, Michigan; and the Fast Flux Test Facility at the Hanford site in Richland, Washington. These activities generated approximately 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel for which the U.S. Department of Energy (DOE) is now responsible. Sodium-bonded spent nuclear fuel is distinguished from commercial nuclear reactor spent nuclear fuel by the presence of metallic sodium, a highly reactive material; frequently by metallic uranium and plutonium, which are also potentially reactive; and in some cases, highly enriched uranium. Metallic sodium in particular presents challenges for management and ultimate disposal of this spent nuclear fuel. For example, metallic sodium reacts with water to produce explosive hydrogen gas and corrosive sodium hydroxide; both could affect operation of a geologic repository.

DOE proposes to resolve this problem by treating and managing the sodium-bonded spent nuclear fuel and facilitating its ultimate disposal in a geologic repository. The reasonable alternatives for this proposed action are determined by the technology options available to DOE. Several technologies that might be used to treat and manage DOE's sodium-bonded spent nuclear fuel are at various stages of development. Among these are: an electrometallurgical treatment process; the plutonium-uranium extraction (PUREX) process; placement of the spent nuclear fuel in high-integrity cans; a melt and dilute process; a glass material oxidation and dissolution system (GMODS) process; a direct plasma arc-vitreous ceramic process; and a chloride volatility process.

The programmatic risk in implementing any of these potential alternatives for treatment and management of sodium-bonded spent nuclear fuel, or of not treating this fuel, is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for placement in a potential geologic repository. While DOE has drafted preliminary waste acceptance criteria for a geologic repository (DOE 1999c), the final acceptance criteria will be more refined. If the proposed repository at Yucca Mountain in Nevada is developed, final acceptance criteria would not be available until about 2005, when the U.S. Nuclear Regulatory Commission (NRC) would issue a construction authorization. Until such time, the preliminary acceptance criteria are intended to be conservative to allow for uncertainties in the performance of engineered and natural barriers and how such performance might impact public and worker health and safety, as well as material isolation.

This environmental impact statement (EIS) follows the June 1, 1995, Record of Decision (60 FR 28680) for the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact*

[1] *The original name of the plant was the Enrico Fermi Atomic Power Plant. The numeral "1" was added to the name in 1969 after Detroit Edison Company began construction of Fermi-2. The plant also is known as Fermi, Fermi-1, or Enrico Fermi-1.*

Statement (Programmatic Spent Nuclear Fuel EIS) (DOE 1995a), in which DOE decided to regionalize spent nuclear fuel management by fuel type for DOE-owned spent nuclear fuel. DOE also decided to: (1) continue environmental restoration activities at the Idaho National Engineering and Environmental Laboratory (INEEL)[2]; (2) develop cost-effective treatment technologies for spent nuclear fuel and waste management; and (3) implement projects and facilities to prepare waste and treat spent nuclear fuel for interim storage and final disposition. The l995 Record of Decision was based partially on the analyses in the Programmatic Spent Nuclear Fuel EIS, which analyzed the potential environmental consequences of alternatives for transporting, receiving, processing, and storing spent nuclear fuel under DOE's responsibility for the next 40 years. The Programmatic Spent Nuclear Fuel EIS also analyzed the consequences of 10 years of waste and spent nuclear fuel management and environmental restoration actions at INEEL.

In addition, DOE committed to remove all spent nuclear fuel from Idaho by 2035 in a 1995 agreement with the State of Idaho (Settlement Agreement and Consent Order [Idaho 1995] issued on October 17, 1995, in the actions of *Public Service Co. of Colorado v. Batt*, No. CV 91-0035-S-EJL [D. Id.], and *United States v. Batt*, No. CV 91-0054-EJL [D. Id.]). Currently, more than 98 percent of DOE's sodium-bonded spent nuclear fuel is located at INEEL near Idaho Falls, Idaho, and is subject to the requirements of this Settlement Agreement and Consent Order. Before sodium-bonded spent nuclear fuel can be removed from the State of Idaho for ultimate disposal, some or all of the fuel may require treatment.

One of the technologies considered for the treatment of sodium-bonded spent nuclear fuel is the electrometallurgical technology. In a 1995 report (NAS 1995), the National Academy of Sciences' National Research Council Committee on Electrometallurgical Techniques for DOE Spent Nuclear Fuel Treatment recommended that DOE confirm the technical feasibility and cost-effectiveness of electrometallurgical treatment of its sodium-bonded spent nuclear fuel. The Council recommended this be done through a technology demonstration using sodium-bonded spent nuclear fuel that had been removed from EBR-II at Argonne National Laboratory-West (ANL-W). Prior to acting on the recommendation, DOE prepared the *Environmental Assessment for the Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West* (DOE 1996a) and issued a Finding of No Significant Impact on May 22, 1996 (61 FR 25647). The Electrometallurgical Treatment Research and Demonstration Project, which began in June 1996, involved the treatment of up to 100 EBR-II driver spent nuclear fuel assemblies and up to 25 EBR-II blanket spent nuclear fuel assemblies (approximately 1.6 metric tons of heavy metal). The driver spent nuclear fuel contained highly enriched uranium and was used in the active region of the nuclear reactor core. The blanket spent nuclear fuel contained depleted uranium and was used in areas around and near the driver spent nuclear fuel in the reactor core. The Electrometallurgical Treatment Research and Demonstration Project was successfully completed in August 1999. The key analytical and experimental results of the demonstration project are provided in the *Spent Fuel Treatment Demonstration Final Report* issued by ANL-W (Benedict et al. 1999) in August 1999. The salient features of the demonstration project and results are discussed in Section 1.6.3.

Parallel to the assessment provided in this *Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (SBSNF EIS), the National Research Council is continuing to evaluate the Electrometallurgical Treatment Research and Demonstration Project. In its most recent report, *Electrometallurgical Techniques for U.S. Department of Energy Spent Fuel Treatment–Spring 1998 Status Report on Argonne National Laboratory's R&D Activity* (NAS 1998), the Council acknowledged progress in the demonstration and recommended that it be carried to completion. Data from the demonstration project were used in preparing this EIS. The National Research Council issued a final report on the technology demonstration in April 2000. DOE will consider the Council's final report in reaching a decision regarding the disposition of sodium-bonded spent nuclear fuel.

[2] *The laboratory's name was changed from Idaho National Engineering Laboratory to Idaho National Engineering and Environmental Laboratory in January 1997.*

1.2 PURPOSE AND NEED FOR ACTION

Sodium-bonded spent nuclear fuel contains metallic sodium. The presence of metallic sodium in the sodium-bonded spent nuclear fuel could potentially complicate disposal certification and licensing for the ultimate disposal of this spent nuclear fuel in a geologic repository. Metallic sodium reacts vigorously with water, producing heat, potentially explosive hydrogen gas, and sodium hydroxide, a corrosive substance. Sodium also is pyrophoric (i.e., a material that is susceptible to spontaneous ignition and continuous combustion). Sodium metal was used as a heat-transfer medium within the stainless steel cladding (outer layer) of the nuclear fuel and as a coolant in the nuclear reactors that used this fuel. To the extent possible, sodium was removed from the external surface of this fuel after its use, but a portion remains bonded to the uranium metal alloy fuel within the cladding and cannot be removed without further treatment. Most (i.e., 99 percent by weight) of the sodium-bonded spent nuclear fuel contains metallic uranium and plutonium. Some metals, such as pure uranium and pure plutonium, are reactive in the presence of air and moisture. The current preliminary repository waste acceptance criteria (DOE 1999c) exclude reactive and potentially explosive materials from being accepted into a geologic repository unless they exist only in trace quantities. Additionally, some of the sodium-bonded spent nuclear fuel contains highly enriched uranium that could create criticality concerns requiring control methods.

To ensure that the terms of the State of Idaho Settlement Agreement and Consent Order are met, and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. Appropriate treatment and management of the sodium-bonded spent nuclear fuel would significantly reduce complications related to disposal qualification. Technologies for spent nuclear fuel treatment that could facilitate such qualification therefore should be considered in reaching a decision for treatment of DOE-owned sodium-bonded spent nuclear fuel. Several treatment technologies are at various stages of development and could be used to remove and stabilize the metallic sodium and immobilize or isolate the transuranic and fission products that are in the sodium-bonded spent nuclear fuel. Such technologies include the electrometallurgical treatment process; the PUREX process; placement of the spent nuclear fuel in high-integrity cans; a melt and dilute process; the GMODS process; a direct plasma arc-vitreous ceramic process; and a chloride volatility process.

It is prudent to evaluate these alternative treatment technologies now, while DOE is performing site characterization activities for a potential geologic repository at Yucca Mountain, Nye County, Nevada. Potential waste forms resulting from treatment or packaging of sodium-bonded spent nuclear fuel should be developed as much as possible in parallel with any repository development. The process of establishing a repository depends on not only the site but also the materials for disposal. As part of this process, a total system performance assessment that describes the probable behavior of the repository is performed. This total system assessment includes the performance of the specific waste forms and inventories proposed for disposal. As part of the process of establishing a repository, data for the waste forms are needed prior to making a final repository selection.

Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in considering the future of PUREX processing capabilities, DOE now needs to decide whether these technologies are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying this NEPA process could result in a loss of capability and of technical staff knowledgeable about and experienced with the demonstration project. This was an important consideration in the decision to proceed with this EIS.

1.3 PUBLIC PARTICIPATION

1.3.1 Issues Identified During the Scoping Period

On February 22, 1999, DOE published in the *Federal Register* a Notice of Intent to prepare an *Environmental Impact Statement for Electrometallurgical Treatment of Sodium-Bonded Spent Nuclear Fuel in the Fuel Conditioning Facility at Argonne National Laboratory-West* (64 FR 8553). In this Notice of Intent, DOE invited the public to participate and comment on the proposed scope of the EIS. Subsequent to this notice, DOE held four public scoping meetings. The first meeting was attended by about 60 persons and was held in Idaho Falls, Idaho, on March 9, 1999. The second meeting was held in Boise, Idaho, on March 11, 1999, and was attended by 7 persons. Ten persons attended the third meeting, which was held in North Augusta, South Carolina, on March 15, 1999. The fourth meeting was held in Arlington, Virginia, on March 18, 1999, and was attended by 8 persons. A court reporter recorded oral comments at each of these meetings. Written statements or comments from the public also were collected at the meetings. In addition, the public was invited to send comments to DOE by letter, e-mail via the Internet, a toll-free telephone number, and facsimile. The public scoping comment period began with the publication of the Notice of Intent in the *Federal Register* on February 22, 1999 (64 FR 8553), and ended 45 days later on April 8, 1999.

A total of 228 comments were received during the public scoping comment period. All comments were reviewed and considered by DOE in developing the scope of this EIS. A summary of scoping comments and their disposition is provided in Appendix A of this EIS. The significant issues identified during the public scoping period are addressed below.

Many commentors at the public scoping meetings asked specific, technical questions about the proposed action. Areas of interest included:

Waste volume reduction
Nature of the spent nuclear fuel at ANL-W
Waste forms characterization
Waste disposition and qualification (repository acceptance criteria)
PUREX process
Use of facilities
Nonproliferation impacts
Transportation
Demonstration project

A number of persons commented on the schedule for this EIS. Many stated that the draft EIS should not be issued for public comment before publication of other related reports, such as the National Research Council's Waste Qualification Assessment and the National Academy of Sciences' Independent Assessment Final Report on the Electrometallurgical Treatment Research and Demonstration Project; a Nonproliferation Impacts Assessment; and an independent Cost Study. Several commentors said that this EIS is premature because the demonstration project will not be completed until after the draft EIS is published.

Several commentors asked that the EIS include information about the costs of the proposed action and all of the technology alternatives under consideration. Other commentors stated that the public should have an opportunity to comment on the Nonproliferation Impacts Assessment in the same time frame as the draft EIS, or that this EIS should be delayed until the Nonproliferation Impacts Assessment becomes publicly available. Some suggested that the Nonproliferation Impacts Assessment be included in the EIS. A few commentors expressed the opinion that electrometallurgical treatment of spent nuclear fuel is a proliferation-prone technology.

Many waste-related comments included opinions about whether low-enriched uranium, plutonium, noble metals, and other components of the waste stream should be viewed as waste or potentially valuable resources. Several commentors asked that the EIS clarify which specific waste forms would be generated by the treatment processes. Others said the EIS should clarify whether the waste would remain at the Savannah River Site (SRS) after processing or be returned to Idaho if the PUREX process were used. Some commentors argued that the electrometallurgical treatment alternative would not reduce the volume of waste to be stored in a repository. A few questioned how DOE can ensure the waste will meet the acceptance criteria for a repository when no one knows what those criteria will be—or if there will be any repository at all. A few others recommended that the EIS evaluate the PUREX process before it is shut down to ensure that the waste forms resulting from electrometallurgical treatment are as good as the borosilicate glass that is being prepared for a geologic repository.

The commentors generally agreed that DOE should evaluate in detail all of the alternative treatment technologies that potentially could meet DOE's treatment and management needs, even those that DOE considers less technologically mature. Several commentors expressed the opinion that DOE already has made a technology decision in favor of electrometallurgical treatment, but that other alternative new technologies should not be dismissed because of a lack of knowledge about them. Some asked that the EIS: (1) explain how DOE can consider the PUREX process a reasonable alternative when, historically, it could not handle sodium-bonded spent nuclear fuel, and (2) evaluate whether changes in the PUREX process would be needed to accommodate sodium-bonded spent nuclear fuel. A few commentors suggested the EIS should analyze blanket and driver spent nuclear fuel separately, since they have different chemical and radiological characteristics and different treatments might be warranted.

Comments concerning environment, safety, and health issues were comparatively few, as were comments about transportation safety and security.

Comments received during the scoping period were systematically reviewed and evaluated to determine whether the issues raised fell within the scope of the EIS. The comments are addressed in the EIS as indicated in Appendix A, Table A–1, which includes references to specific EIS sections. As a result of public comment, DOE changed the proposed action of the EIS, as well as the structure of the alternatives. The proposed action was changed from electrometallurgical treatment of sodium-bonded spent nuclear fuel in the Fuel Conditioning Facility at ANL-W to the treatment and management of sodium-bonded spent nuclear fuel. The title also was changed accordingly. This change was made to alleviate concerns about bias for one treatment technology over others. The alternatives were restructured to reflect differences in the characteristics of the different types of sodium-bonded spent nuclear fuel. Thus, several alternatives were added that treat driver and blanket spent nuclear fuel by different technologies.

Issues related to cost and nuclear nonproliferation were not considered to be within the scope of the EIS. However, DOE conducted a Cost Study and a Nonproliferation Impacts Assessment for the reasonable alternatives. These reports were made available to the public during the public review process.

With respect to comments related to the ongoing Electrometallurgical Treatment Research and Demonstration Project, data from the project were used for the preparation of both the draft and the final EIS as indicated in Section 1.6.3.

Comments considered to be not within the scope of the EIS are listed in Appendix A, Table A–3, along with an explanation for their disposition.

1.3.2 Issues Raised During the Public Comment Period on the Draft EIS

In July 1999, DOE published the *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*. The regulations implementing the National Environmental Policy Act (NEPA) mandate a minimum 45-day public comment period after publication of a draft EIS to provide an opportunity for the public and other stakeholders to comment on the EIS analysis and results. The 45-day public comment period on the SBSNF Draft EIS began on July 31, 1999, and was scheduled to end on September 13, 1999. In response to commentor requests, the comment period was extended an additional 15 days through September 28, 1999. During this 60-day comment period, public hearings were held in North Augusta, South Carolina; Boise and Idaho Falls, Idaho; and Arlington, Virginia. In addition, the public was encouraged to submit comments via the U.S. mail service, electronic mail, a toll-free 800-number phone line, and a toll-free fax line.

A total of 494 comments were received during the public comment period. Most of the comments focused on the following issues: (1) the purpose, need for, and timing of the proposed action; (2) the introduction of new waste forms produced by the proposed action, their acceptability in a geologic repository, and the disposition of uranium and plutonium by-products; (3) the public availability of information considered relevant to reviewing the draft EIS, the extension of the comment period, and the relationship of the EIS to other DOE programs; (4) the cost of the various alternatives; (5) the impacts of the proposed action on U.S. nuclear nonproliferation policy; (6) technical or NEPA-related questions regarding technologies and alternatives; and (7) questions related to the affected environment and the environmental consequences. DOE's responses to these issues are summarized below. The comments also dealt with a number of other subjects, including technologies considered and dismissed from further evaluation, long-term (beyond institutional control) performance of the sodium-bonded spent nuclear fuel during storage on site, and questions on the methodology and assumptions of the health and safety analysis. Many commentors expressed their opposition or support for DOE's action in general or for specific alternatives under the proposed action or the No Action Alternative. Section A.2 of Appendix A provides DOE's responses to all comments on a comment-by-comment basis.

Purpose, Need for, and Timing of the Proposed Action

Many comments expressed the opinion that DOE failed to demonstrate the purpose and need for the proposed action or to provide a rationale for its timing. Some of the reasons given included the lack of a compelling argument that there is a safety risk associated with current storage; the lack of a regulatory framework and final waste acceptance criteria; the lack of an approved site for a geologic repository; insufficient information on the results of the Electrometallurgical Treatment Research and Demonstration Project; and the lack of analysis showing that direct disposal of the sodium-bonded spent nuclear fuel without sodium removal would be detrimental to the performance of the geologic repository.

DOE's position, presented in the EIS, is that the need to examine options for the treatment and management of sodium-bonded spent nuclear fuel is based on the existing regulatory environment concerning long-term disposal of spent nuclear fuel and high-level radioactive waste. DOE assumes that its sodium-bonded spent nuclear fuel, as well as other DOE-owned spent nuclear fuel, eventually will be disposed of in a geologic repository. However, one of the key requirements, as specified in the current April 1999 version of DOE's Waste Acceptance Systems Requirements Document (DOE 1999c) and in NRC requirements for acceptance of spent nuclear fuel or high-level radioactive waste in a geologic repository, is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective [10 CFR 60.135(b)(1)]. The sodium-bonded spent nuclear fuel, if left in its existing state, would contain pyrophoric and chemically reactive metallic sodium and therefore may not meet DOE or NRC repository acceptance criteria, or would complicate the qualification process.

The timing for the proposed action is a programmatic issue rather than a safety issue. The EIS does not conclude that current storage of sodium-bonded spent nuclear fuel presents a threat to the health and safety of workers or the public. The programmatic risk associated with implementing the proposed action or not treating the sodium-bonded spent nuclear fuel is the uncertainty surrounding the acceptability of this fuel for placement in a geologic repository. The process of establishing a repository is dependent on not only the site but also the materials to be disposed. As part of this process, a total system performance assessment that describes the probable behavior of a repository is performed. This total system assessment includes the performance of the specific waste forms and inventories proposed for disposal. As part of the process of establishing a repository, data for the waste forms are needed prior to making a final repository selection, not after. In fact, if specific waste forms are not represented in crucial documents like this EIS, new documentation will be needed to allow the possibility of disposing of those materials in the repository. The performance of sodium-bonded spent nuclear fuel in a geologic repository depends on many factors (e.g., long-term fuel integrity and fuel/waste package survivability in a repository environment), and the presence of metallic sodium would complicate the modeling even further. Stabilization of the sodium-bonded spent nuclear fuel and/or removal of the metallic sodium would provide greater protection for human health and the environment.

The Electrometallurgical Treatment Research and Demonstration Project began in June 1996 and, although the review of the test results has not been finalized in a single report, a number of status reports were issued by DOE and reviewed by the National Academy of Sciences' National Research Council Committee. They are referenced in the EIS. The success criteria established at the outset of the project have been fulfilled. The environmental impact analysis associated with the electrometallurgical treatment process alternatives was based on actual data from the demonstration project. This final EIS includes a new section on the status and results of the project. Having completed the demonstration project and in considering the future of its PUREX processing capabilities, DOE now needs to decide whether these technologies are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the NEPA process could result in a loss of capability and of technical staff knowledgeable about and experienced with the demonstration project. This was an important consideration in the decision to proceed with this EIS.

New Waste Forms and Disposition of Uranium and Plutonium By-Products

Some of the comments questioned the generation of new waste forms from treating the sodium-bonded spent nuclear fuel and the possible acceptance of these forms in a geologic repository. Also, a number of commentors remarked on the generation of uranium and plutonium as by-products of the treatment process. Related issues were the disposition of uranium metal, a by-product of the electrometallurgical process, and the compliance of both the PUREX and the electrometallurgical processes with U.S. nuclear nonproliferation policy in terms of the separation of these elements.

All of the alternatives evaluated in this EIS would produce some form of high-level radioactive waste. Electrometallurgical treatment would produce two new waste forms (i.e., metallic and ceramic) and the melt and dilute process would produce a new metallic form (i.e., a melt and dilute product, or conditioned spent nuclear fuel). These forms would be more stable than the untreated sodium-bonded spent nuclear fuel. The production of a chemically stable waste form to replace a chemically reactive waste form (i.e., sodium-bonded spent nuclear fuel) represents an improvement in the safe, long-term storage of this spent nuclear fuel. DOE expects the new waste forms to be suitable for disposal in a repository and to meet the requirements of the final waste acceptance criteria. The high-level radioactive waste form resulting from the PUREX process is borosilicate glass, which has been tested and analyzed extensively under conditions relevant to a geologic repository.

With respect to uranium and plutonium disposition, the EIS states that any uranium that would be separated under the electrometallurgical process would be blended down and stored on site if it originates from driver spent nuclear fuel, or would be stored on site as depleted uranium if it originates from blanket spent nuclear fuel. The final disposition of the stored uranium has not been decided and is not discussed in the EIS. The disposition of the uranium will be subject to a separate NEPA review. The nuclear nonproliferation policy aspects of this separation are subject to the nuclear nonproliferation policy assessment of the alternatives. The approximately 260 kilograms (572 pounds) of plutonium that would be separated under the PUREX process would be disposed of in accordance with the Record of Decision (65 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999e) issued in November 1999. This separation is the subject of the Nonproliferation Impacts Assessment, which is independent of this EIS.

Public Availability of Information and Related Documentation

Many commentors asked for a 60-day extension of the 45-day public comment period on the draft EIS. Commentors said they wanted additional time to obtain and review relevant documents such as the Yucca Mountain Draft EIS and the National Academy of Sciences' National Research Council's final report on the Electrometallurgical Treatment Research and Demonstration Project, as well as the Cost Study and the Nonproliferation Impacts Assessment. The comments frequently stated that DOE needs to make all of this information publicly available before the end of the EIS comment period and the issuance of the final EIS and the Record of Decision.

In an effort to ensure that all interested parties had time to comment on the draft EIS, the due date for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). With respect to the need for more information, DOE made that information available to the public. Background materials were placed in public reading rooms and were made available to the public through a series of hearings held August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. Materials placed in the reading rooms included the electrometallurgical demonstration environmental assessment, the Finding of No Significant Impact for the environmental assessment, National Research Council reports, the 1995 Settlement Agreement and Consent Order with the State of Idaho, the scoping meeting transcripts and comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS. Although these reports are not critical to the evaluation of the analysis presented in the draft EIS, they will provide input to the Record of Decision. While the final National Research Council report on the demonstration project was published in April 2000, interim status reports were produced throughout the project. Data generated during the demonstration project were used in preparing the EIS.

Cost Issues

A number of commentors raised cost issues and provided comments directly related to the Cost Study, which is not part of the EIS.

Comments concerning the costs of the proposed action were considered beyond the scope of the EIS. The EIS was prepared in accordance with NEPA, as well as the Council on Environmental Quality's regulations on implementing NEPA (40 CFR 1500 through 1508) and DOE's NEPA regulations (10 CFR 1021). None of these regulations require the inclusion of a cost analysis in an EIS. The basic objective of the SBSNF EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for treating and managing sodium-bonded spent nuclear fuel and information about their potential impacts on

public health and safety and the environment. While cost could be an important factor in the ultimate Record of Decision, the purpose of this EIS is to address the environmental consequences of all alternatives under the proposed action and the No Action Alternative. DOE distributed cost information through the independent Cost Study released in August 1999, and this information is available to the public on request and in the DOE public reading rooms. Responses to specific comments related to cost issues are included in Section A.2 of Appendix A.

Nuclear Nonproliferation Policy Issues

The nuclear nonproliferation implications of the proposed action were the subject of a number of comments. Some commentors expressed strong opinions about how the use of specific technologies such as electrometallurgical treatment might impact U.S. nonproliferation policy.

Nonproliferation is another issue that was considered beyond the scope of the EIS. A separate Nonproliferation Impacts Assessment was prepared by DOE's Office of Arms Control and Nonproliferation. After assessing the potential nonproliferation impacts that could result from each of the alternatives and technologies analyzed in the SBSNF Draft EIS, the Office of Arms Control and Nonproliferation found that all the alternatives, except that involving PUREX processing at SRS, are fully consistent with U.S. policy concerning reprocessing and nuclear nonproliferation. Electrometallurgical treatment, for example, would not increase national inventories of weapons-usable fissile material because, although highly enriched uranium is an interim product of the process, it would be blended down to low-enriched uranium during treatment. Within the current equipment configuration and design, it is not possible to produce weapons-usable plutonium merely by adjusting the operating parameters. To do this, traditional aqueous processing would be required after electrometallurgical treatment. However, traditional aqueous processing could be used to produce weapons-usable plutonium directly from the spent nuclear fuel, without electrometallurgical treatment, so electrometallurgical treatment itself does not present a special proliferation concern. Responses to specific comments related to nonproliferation are included in Section A.2 of Appendix A.

Technologies, Alternatives

Various comments dealt with technical questions and issues regarding the treatment technologies addressed in the EIS or NEPA-related issues regarding the selected alternatives.

The variety of the issues precludes a summary response. Responses to these questions on a comment-by-comment basis are included in Section A.2 of Appendix A. A number of revisions to the EIS were made as a result of these comments.

Affected Environment and Consequences

A number of comments included questions concerning the description of the affected environment in the SBSNF Draft EIS, and the results of the environmental impact analysis.

As in the case above, responses to these questions on a comment-by-comment basis are included in Section A.2 of Appendix A.

1.4 SCOPE OF THIS EIS

The EIS evaluates the potential direct, indirect, and cumulative environmental impacts associated with the treatment of sodium-bonded spent nuclear fuel in one or more spent nuclear fuel management facilities. In addition, this EIS evaluates the environmental impacts of the No Action Alternative.

DOE proposes to treat and manage sodium-bonded spent nuclear fuel at one or more of the following spent nuclear fuel management facilities: ANL-W at INEEL and the F-Canyon or Building 105-L at SRS. The impacts from the treatment and management of sodium-bonded spent nuclear fuel at INEEL and SRS and their spent nuclear fuel management facilities are described in this EIS. In addition to the No Action Alternative, the EIS analyzes six reasonable alternatives under the proposed action that employ one or more of the following technology options: electrometallurgical treatment, the PUREX process, packaging in high-integrity cans, and the melt and dilute treatment process. Electrometallurgical treatment at a site other than ANL-W, the GMODS process, the direct plasma arc-vitreous ceramic treatment, and the chloride volatility process were considered and deemed not to be reasonable alternatives under the proposed action.

This EIS analyzes the potential environmental impacts associated with the proposed action, which includes: (1) preparation prior to treatment; (2) treatment and management; (3) transportation; and (4) decontamination and deactivation of equipment that would be installed for the purpose of implementing a specific treatment method. Impacts from the transport to INEEL of sodium-bonded spent nuclear fuel from DOE sites such as the Hanford site in Washington, Sandia National Laboratories in New Mexico, and Oak Ridge National Laboratory in Tennessee are addressed in the Programmatic Spent Nuclear Fuel EIS (DOE 1995a).

The United States does not encourage the civilian use of plutonium and, accordingly, does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes. However, two of the technologies under the proposed action involve the separation of plutonium (PUREX) and highly enriched uranium (electrometallurgical treatment). To address concerns that treatment of this fuel by chemical separation could encourage reprocessing in other countries, DOE's Office of Nonproliferation and National Security independently evaluated the impacts of each treatment technology on U.S. nonproliferation efforts. The Nonproliferation Impacts Assessment was published at about the same time as the draft EIS.

1.5 DECISIONS TO BE MADE

Based on the analytical results of this EIS as well as cost, schedule, and nonproliferation considerations, DOE intends to make the following decisions:

- Whether to use an existing, mature technology to treat the sodium-bonded spent nuclear fuel, and if so, which technology should be selected and where it should be implemented.

- Whether to take no action now and wait for further information regarding the potential development of a geologic repository, or promote the development of a less mature or new treatment technology.

The information presented in this EIS, combined with public comments on the draft EIS, the Nonproliferation Impacts Assessment, a Cost Study of the reasonable alternatives, and the National Research Council's final evaluation of the demonstration project, will enable DOE to make a decision regarding treatment and management of the sodium-bonded spent nuclear fuel.

1.6 RELATIONSHIP TO OTHER ACTIONS AND PROGRAMS

This section explains the relationship between this EIS and other relevant NEPA documents and programs. Completed NEPA actions are described in Section 1.6.1, ongoing actions are described in Section 1.6.2, and the Electrometallurgical Treatment Research and Demonstration Project is described in Section 1.6.3.

1.6.1 Completed NEPA Actions

1.6.1.1 Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement

This Programmatic Spent Nuclear Fuel EIS (DOE 1995a) analyzed at a programmatic level the potential environmental consequences of alternatives used for 40 years to transport, receive, process, and store spent nuclear fuel under DOE's responsibility. It also analyzed the consequences of 10 years of waste and spent nuclear fuel management and environmental restoration actions at Idaho National Engineering Laboratory (now known as INEEL). For programmatic spent nuclear fuel management, this document analyzed alternatives that included no action, decentralization, regionalization, centralization, and the use of plans that existed in 1992 and 1993 for the management of these materials. For the INEEL, this document analyzed alternatives such as no action, a 10-year plan, and minimum and maximum treatment, storage, and disposal of DOE waste.

Issued in April 1995, the Programmatic Spent Nuclear Fuel EIS was followed by a Record of Decision published in the *Federal Register* on June 1, 1995 (60 FR 28680). In the Record of Decision, DOE decided to regionalize spent nuclear fuel management by fuel type for DOE-owned spent nuclear fuel. DOE also decided to: (1) continue environmental restoration activities at the INEEL; (2) develop cost-effective treatment technologies for spent nuclear fuel and waste management; and (3) implement projects and facilities to prepare waste and treat spent nuclear fuel for interim storage and final disposition. The SBSNF EIS was prepared as a follow-on to this programmatic EIS.

The June 1, 1995, Record of Decision was later amended to reflect the October 16, 1995, Settlement Agreement and Consent Order between DOE, the State of Idaho, and the U.S. Department of the Navy pertaining to spent nuclear fuel shipments into and out of the State of Idaho. The amendment to the Record of Decision was published in the *Federal Register* on March 8, 1996 (61 FR 9441). In this amendment, DOE did not modify or rescind any of the provisions presented in the June 1, 1995, Record of Decision (60 FR 28680), but reduced the number of shipments of spent nuclear fuel into the State of Idaho.

1.6.1.2 Savannah River Site Waste Management Final Environmental Impact Statement

DOE issued this EIS (DOE 1995b) to provide a basis for the selection of a site-wide approach to managing present and future (through 2024) waste generated at SRS. This waste would come from ongoing operations and potential actions, new missions, environmental restoration, and decontamination and decommissioning programs.

The SRS Waste Management EIS 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 radioactive waste evaporator in the H-Area tank farm. In addition, it evaluates the Consolidated Incineration Facility for the treatment of mixed waste. The Record of Decision, published in the *Federal Register* on October 30, 1995 (60 FR 55249), stated that DOE will configure its waste management system according to the moderate treatment alternative described in the EIS. The SRS Waste Management EIS evaluates management alternatives for various types of waste that actions proposed in this EIS could generate.

In a Supplemental Record of Decision published in the *Federal Register* on May 19, 1997 (62 FR 27241), DOE decided to take additional measures to further implement the Moderate Treatment Configuration Alternative for mixed waste and transuranic waste. This decision was based on the SRS Waste Management EIS and was consistent with completed negotiations between DOE and the State of South Carolina.

1.6.1.3 Final Environmental Impact Statement, Interim Management of Nuclear Materials

In this EIS (DOE 1995c) DOE evaluated actions to stabilize nuclear materials at SRS that present potential environmental, safety, and health risks in their current storage condition or may present a risk within the next 10 years. As a result, DOE published five decisions from this EIS. In the Record of Decision, published in the *Federal Register* on December 19, 1995 (60 FR 65300), DOE decided to process, blend, and/or vitrify specific amounts of plutonium, uranium, americium, and curium solutions, and spent nuclear fuel down to low enrichments and/or some other form of stable material. The Savannah River Site Interim Management of Nuclear Materials EIS evaluates the treatment and management of spent nuclear fuel and other waste at SRS such as those generated by the proposed actions in the SBSNF EIS.

In the first, second, and third supplements to the Record of Decision, published in the *Federal Register* on February 21, 1996; September 13, 1996; and April 11, 1997, respectively (61 FR 6633, 61 FR 48474, and 62 FR 17790), DOE decided to stabilize additional amounts of spent nuclear fuel and other materials by processing them in the F-Canyon, H-Canyon, and the FB-Line and blending the resulting highly enriched uranium down to low-enriched uranium. DOE then would transfer the resulting nuclear material to the SRS high-level radioactive waste tanks for vitrification in the Defense Waste Processing Facility.

In the fourth supplement to the Record of Decision, published in the *Federal Register* on November 14, 1997 (62 FR 61099), DOE decided to process, store, and vitrify specific amounts of nuclear material in the Defense Waste Processing Facility and to amend the September 13, 1996, supplement to the Record of Decision (61 FR 48474) to address additional amounts of plutonium and neptunium solutions stored at SRS.

1.6.1.4 Environmental Assessment for the Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West

This NEPA analysis (DOE 1996a) addressed the environmental impacts associated with a research and demonstration project involving the electrometallurgical treatment of up to 100 EBR-II driver spent nuclear fuel assemblies and up to 25 EBR-II blanket spent nuclear fuel assemblies in the Fuel Conditioning Facility at ANL-W. As noted in the environmental assessment, DOE had identified electrometallurgical treatment as a promising technology to treat EBR-II spent nuclear fuel, but an appropriate demonstration was needed to provide DOE with sufficient information to evaluate its technical feasibility. A successful demonstration of the electrometallurgical treatment technology on EBR-II spent nuclear fuel, combined with research and testing of the resulting waste forms, would provide DOE with the information needed to determine whether this treatment technology should be used to treat the remainder of EBR-II spent nuclear fuel and/or other types of spent nuclear fuel. Based on the analysis presented in the environmental assessment, and after consideration of all the comments received from the public, DOE decided to proceed with the proposed demonstration and finalized the environmental assessment on May 15, 1995. DOE also determined that the proposed action did not constitute a major Federal action and would not necessitate the preparation of an EIS. DOE issued a Finding of No Significant Impact, which was published in the *Federal Register* on May 22, 1996 (61 FR 25647).

The electrometallurgical treatment process that was addressed in this environmental assessment is basically the same process that is being evaluated in this EIS. The process involves the dissolution of spent nuclear fuel by the use of an electric current in a molten salt mixture. The only difference between the environmental assessment and this SBSNF EIS is the amount of spent nuclear fuel being considered for treatment.

The Electrometallurgical Treatment Research and Demonstration Project was completed in August 1999. Salient features of the project and results are discussed in Section 1.6.3.

1.6.1.5 Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement

DOE prepared this EIS (DOE 1996b) because of the need to move rapidly to neutralize the proliferation threat of surplus highly enriched uranium and to demonstrate to other nations the United States' commitment to nonproliferation. The Highly Enriched Uranium EIS evaluates management alternatives for materials that actions proposed in this EIS could generate.

In the Record of Decision, published in the *Federal Register* on August 5, 1996 (61 FR 40619), DOE stated it would implement a program that will gradually blend as much as 85 percent of the surplus highly enriched uranium to a uranium-235 enrichment level of approximately 4 percent, and will blend the remaining surplus highly enriched uranium down to an enrichment level of about 0.9 percent for disposal as low-level radioactive waste. This will occur over 15 to 20 years. DOE could use different technologies at four potential blending facilities, including SRS and the Oak Ridge Reservation. Blending down highly enriched uranium would affect SRS operations and waste generation.

1.6.1.6 Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste

This Final Waste Management Programmatic EIS (DOE 1997) examined the potential environmental and cost impacts of strategic management alternatives for managing five types of radioactive and hazardous waste that have resulted and will continue to result from nuclear defense and research activities at a variety of sites around the United States. The five waste types are mixed waste, low-level radioactive waste, transuranic waste, high-level radioactive waste, and hazardous waste. This programmatic EIS provided information on the impacts of various siting alternatives which DOE will use to decide at which sites to locate additional treatment, storage, and disposal capacity for each waste type. This information included the cumulative impacts of combining future siting configurations for the five waste types and the collective impacts of other past, present, and reasonably foreseeable future activities. The programmatic EIS evaluates management and treatment alternatives for various types of waste that actions proposed in this EIS could generate.

The waste management facilities considered for the five waste types were treatment and disposal facilities for mixed waste; treatment and disposal facilities for low-level radioactive waste; treatment and storage facilities for transuranic waste in the event that treatment is required before disposal; storage facilities for treated (vitrified) high-level radioactive waste canisters; and treatment of nonwastewater hazardous waste by DOE and commercial vendors. In addition to the No Action Alternative, which included only existing or approved waste management facilities, the alternatives for each of the five waste type configurations included decentralized, regionalized, and centralized alternatives for operating existing and new waste management facilities. However, the siting, construction, and operation of any new facility at a selected site would not be decided until completion of a site-wide or project-specific environmental review.

DOE has published four decisions from this programmatic EIS. In the first Record of Decision, published in the *Federal Register* on January 23, 1998 (63 FR 3629), DOE decided that each DOE site that currently has or will generate transuranic waste will prepare and store its transuranic waste on site, except for Sandia National Laboratories/New Mexico, which will transfer its transuranic waste to the Los Alamos National Laboratory. Los Alamos National Laboratory will have facilities that are not available or anticipated at Sandia National Laboratories to prepare and store transuranic waste prior to disposal.

In the second Record of Decision, published in the *Federal Register* on August 5, 1998 (63 FR 41810), DOE decided to continue using offsite facilities for the treatment of major portions of the nonwastewater hazardous waste generated at DOE sites. This decision did not involve any transfer of nonwastewater hazardous waste among DOE sites.

In the third Record of Decision, published in the *Federal Register* on August 26, 1999 (64 FR 46661), DOE decided to store immobilized high-level radioactive waste in a final form at the site of generation (Hanford, INEEL, SRS, and the West Valley Demonstration Project) until transfer to a geologic repository for ultimate disposal.

DOE addressed the management and disposal of low-level and mixed radioactive waste in a fourth Record of Decision, published in the *Federal Register* on February 25, 2000 (65 FR 10061). In this Record of Decision, DOE decided to perform minimum treatment of low-level radioactive waste at all sites and continue, to the extent practicable, disposal of onsite low-level radioactive waste at INEEL, Los Alamos National Laboratory, the Oak Ridge Reservation, and SRS. DOE decided to treat mixed low-level radioactive waste at the Hanford site, INEEL, the Oak Ridge Reservation, and SRS, with disposal at the Hanford site and the Nevada Test Site.

1.6.1.7 Advanced Mixed Waste Treatment Project Final Environmental Impact Statement

This EIS (DOE 1999a) assessed the potential environmental impacts associated with four alternatives related to the construction and operation of the Advanced Mixed Waste Treatment Facility at INEEL. The alternatives analyzed were: the No Action Alternative; the proposed action; the Nonthermal Treatment Alternative; and the Treatment and Storage Alternative. The Advanced Mixed Waste Treatment Facility would treat transuranic waste, mixed waste, and alpha-contaminated mixed waste in preparation for disposal. After treatment, transuranic waste would be disposed of at the Waste Isolation Pilot Plant in New Mexico. Mixed waste would be disposed of at an approved disposal facility depending on decisions to be based on DOE's Final Waste Management Programmatic EIS (DOE 1997). Evaluations of impacts on land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air resources; water resources; ecological resources; noise; traffic and transportation; occupational and public health and safety; INEEL services; and environmental justice were included in the assessment. The *Advanced Mixed Waste Treatment Project Final Environmental Impact Statement* addresses waste types that could be generated by actions proposed in this EIS.

In the Record of Decision, published in the *Federal Register* on April 7, 1999 (66 FR 16948), DOE decided to proceed with the construction and operation of the Advanced Mixed Waste Treatment Facility. DOE then will treat and prepare for shipment and disposal 65,000 cubic meters (2.3 million cubic feet) of DOE transuranic waste, mixed waste, alpha-contaminated mixed waste currently stored at INEEL. As a result of the decision to complete this facility, DOE also could treat up to 120,000 cubic meters (4.24 million cubic feet) of additional waste from INEEL or other DOE sites for a total of 185,000 cubic meters (6.53 million cubic feet). The Advanced Mixed Waste Treatment Facility will treat waste to meet the Waste Isolation Pilot Plant Waste Acceptance Criteria and applicable requirements of the Toxic Substances Control Act and the Resource Conservation and Recovery Act (RCRA) Land Disposal Restrictions.

In makings its decision, DOE considered several factors, including the environmental analyses reported in the Advanced Mixed Waste Treatment Project Final EIS; estimated costs of the alternatives reported in the Advanced Mixed Waste Treatment Project EIS Alternatives Cost Study; regulatory implications of the alternatives; mission; national policy; and public comments on the Advanced Mixed Waste Treatment Project Draft EIS. This Record of Decision (66 FR 16948) documents DOE's decision to implement the Preferred Alternative, which provides the greatest long-term protection of the environment with small short-term environmental impacts and health risks.

1.6.1.8 Surplus Plutonium Disposition Final Environmental Impact Statement

The Surplus Plutonium Disposition EIS (DOE 1999e) was tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic Environmental Impact Statement* (DOE 1996c). The Record of Decision for the programmatic EIS, published in the *Federal Register* on January 14, 1997

(62 FR 3014), outlined DOE's approach to plutonium disposition and established the groundwork for the Surplus Plutonium Disposition EIS. The fundamental purpose of the program is to ensure that plutonium produced for nuclear weapons and declared excess to national security needs (now and in the future) will never again be used for nuclear weapons.

The Surplus Plutonium Disposition EIS evaluated reasonable alternatives for the siting, construction, and operation of facilities required to implement DOE's disposition strategy for up to 50 metric tons of surplus plutonium, including a No Action Alternative. The disposition facilities analyzed in this EIS include pit disassembly and conversion, plutonium conversion and immobilization, and mixed oxide fuel fabrication. The Surplus Plutonium Disposition EIS also analyzed the potential impacts of fabricating a limited number of mixed oxide fuel assemblies for testing in a reactor. The Surplus Plutonium Disposition EIS is a related NEPA action because it addresses the disposition of material that the SBSNF EIS could generate.

In the Record of Decision, published in the *Federal Register* on January 11, 2000 (65 FR 1608), DOE decided to provide for the safe and secure disposition of up to 50 metric tons of surplus plutonium by constructing all three disposition facilities, pit disassembly and conversion, plutonium conversion and immobilization, and mixed oxide fuel fabrication, at SRS. DOE also decided to implement the mixed oxide fuel alternative analyzed in the EIS.

1.6.2 Ongoing NEPA Actions

1.6.2.1 Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement

The *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000) was issued in March 2000. This SRS Spent Nuclear Fuel Final EIS analyzed the potential impacts from the management of spent nuclear fuel and targets assigned to SRS, including placing these materials in forms suitable for ultimate disposition. Options to treat, package, and store spent nuclear fuel are discussed in this document. The material addressed by this EIS consists of approximately 68 metric tons of heavy metal of spent nuclear fuel (including 20 metric tons of heavy metal of uranium-thorium spent nuclear fuel at SRS; approximately 28 metric tons of heavy metal of aluminum-clad spent nuclear fuel from foreign and domestic research reactors to be shipped to SRS through 2035; and 20 metric tons of heavy metal of stainless steel or zirconium-clad spent nuclear fuel, as well as some other programmatic material stored at SRS for repackaging and dry storage pending shipment off site).

The alternatives considered in the SRS Spent Nuclear Fuel Final EIS encompass a range of new packaging, new processing, and conventional reprocessing technologies for the treatment of spent nuclear fuel. Many of these technologies also are analyzed in this SBSNF EIS. However, in the SRS Spent Nuclear Fuel Final EIS, DOE chose melt and dilute and conventional processing (PUREX) as preferred treatment alternatives for the spent nuclear fuel assigned to SRS.

1.6.2.2 Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada

This draft EIS (DOE 1999b) assesses the potential environmental impacts from the proposed construction, operation, monitoring, and closure of an NRC-licensed geologic repository for the disposal of spent nuclear fuel and high-level radioactive waste, as mandated by the Nuclear Waste Policy Act, as amended. The Yucca Mountain EIS is required to accompany any DOE site recommendation to the President, as appropriate, under Section 114 of the Nuclear Waste Policy Act.

The proposed action addressed in this EIS is to construct, operate and monitor, and eventually close a geologic repository at Yucca Mountain in southern Nevada for the disposal of spent nuclear fuel and high-

level radioactive waste currently in storage at 72 commercial and 5 DOE sites across the United States. The EIS evaluates (1) projected impacts on the Yucca Mountain environment from the construction, operation and monitoring, and eventual closure of the geologic repository; (2) the potential long-term impacts of repository disposal of spent nuclear fuel and high-level radioactive waste; (3) the potential impacts of transporting these materials nationally and in the state of Nevada; and (4) the potential impacts of not proceeding with the proposed action. Included in the high-level radioactive waste that is assumed to be disposed of at the repository are the metallic and ceramic waste forms that would be produced by the electrometallurgical treatment of both driver and blanket sodium-bonded spent nuclear fuel.

Under the No Action Alternative, the EIS evaluates the potential impacts of the continued storage of spent nuclear fuel and high-level radioactive waste at the current storage locations using two scenarios: the first assumes continued storage under institutional controls for at least 10,000 years, and the second assumes no institutional controls after 100 years.

The SBSNF EIS considers the potential disposal at a geologic repository of spent nuclear fuel or high-level radioactive waste that may result from the proposed action involving sodium-bonded spent nuclear fuel. The Yucca Mountain Draft EIS includes the potential long-term impacts of repository disposal from electrometallurgically treated sodium-bonded spent nuclear fuel using data presented in the SBSNF Draft EIS. Quantities of radioactive waste analyzed in the Yucca Mountain document were based on previous projections that have been updated in this document.

1.6.2.3 Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement

This draft EIS was issued in December 1999 (DOE 1999d). It evaluates alternatives for managing the high-level radioactive waste and associated radioactive waste and facilities at INEEL. Under the terms of the 1995 Settlement Agreement and Consent Order with the State of Idaho, DOE agreed to treat high-level radioactive waste currently stored at INEEL and to prepare the waste in a form ready to be shipped out of the State of Idaho by 2035. The purpose of this EIS is to assist DOE in making decisions concerning the management of this radioactive waste to ensure compliance with applicable laws and regulations, and protect the environment and the health and safety of the workers and the public in a cost-effective manner. The high-level radioactive waste generated by the treatment and management of sodium-bonded spent nuclear fuel at ANL-W would not require any additional treatment at INEEL and is not evaluated in the Idaho High-Level Waste EIS.

In this EIS, DOE evaluates reasonable alternatives and options for the treatment of high-level radioactive waste, sodium-bearing waste, newly generated waste, and the disposition of facilities associated with high-level radioactive waste generation, treatment, and storage at INEEL. In addition, this EIS is integrated with the ongoing Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) program at the Idaho Nuclear Technology and Engineering Center. The proposed action under this EIS contributes to the cumulative impacts at the site discussed in the SBSNF EIS.

1.6.3 Electrometallurgical Treatment Research and Demonstration Project

Before electrometallurgical treatment could be considered as a technology choice for treating EBR-II spent nuclear fuel, an appropriate demonstration project was needed to evaluate its technical feasibility. As a preliminary step to demonstration, DOE requested that the National Academy of Sciences' National Research Council conduct an independent assessment of electrometallurgical treatment technology and its potential application to EBR-II spent nuclear fuel. In its report, published in 1995, the National Research Council recommended DOE proceed with demonstrating the technical feasibility of electrometallurgical treatment using a fraction of the EBR-II spent nuclear fuel. Following the National Research Council's recommendation, DOE conducted an environmental assessment of the demonstration project. The environmental assessment was

completed in May 1996 and resulted in a Finding of No Significant Impact, so that no further NEPA review was necessary for the demonstration project to proceed (Benedict et al. 1999).

In June 1996, DOE initiated a three-year testing program at ANL-W to demonstrate the technical feasibility of electrometallurgical treatment of up to 100 EBR-II driver spent nuclear fuel assemblies and up to 25 depleted uranium EBR-II blanket spent nuclear fuel assemblies. These two types of EBR-II spent nuclear fuel, driver and blanket, are typical of most of DOE's sodium-bonded spent nuclear fuel inventory (Benedict et al. 1999). The number of driver spent nuclear fuel assemblies was selected to provide the minimum fission product loading (approximately 3 percent) needed to evaluate the effectiveness of the removal of fission products from the electrorefiner salt and their concentration in the ceramic waste form. The blanket spent nuclear fuel assemblies were treated using a high-throughput electrorefiner that was installed in ANL-W's Fuel Conditioning Facility to evaluate higher-efficiency electrorefining (DOE 1996a).

A total of 100 driver spent nuclear fuel assemblies were treated. These assemblies required multiple batch operations of the treatment equipment in a remote, radioactive hot cell with an inert argon atmosphere. These operations were considered sufficient to demonstrate a dependable, predictable process, including uptime, repair and maintenance, and the operability of the linked process steps. A repeatability demonstration was completed by processing 12 driver spent nuclear fuel assemblies under the same processing conditions. In addition, processing 100 driver spent nuclear fuel assemblies dissolved sufficient active fission products in the electrorefiner salt so that ceramic waste form samples could be produced with representative waste loadings. The purpose of including blanket spent nuclear fuel assemblies in the test program was to demonstrate the mass throughput capacity of the process equipment and facility. A one-month throughput test was completed and a total of 13 blanket spent nuclear fuel assemblies were treated by the end of August 1999, when the demonstration project was concluded (Benedict et al. 1999).

To support the Electrometallurgical Treatment Research and Demonstration Project, DOE established an extensive research and development program at Argonne National Laboratory-East. The largest element of this research and development program involved development, testing, and qualification of the ceramic waste form. Another element was experimental support for electrorefining and metal processing operations in the Fuel Conditioning Facility. In addition, the research and development program included a modeling activity aimed at understanding and improving the electrometallurgical treatment process as well as laying out the requirements for production-scale treatment of the remaining EBR-II spent nuclear fuel. The combined results of the research and development program at Argonne National Laboratory-East and the spent fuel treatment operations at ANL-W provided the technical basis for final evaluation of the electrometallurgical treatment process. An extensive series of topical reports was prepared to present the results of the demonstration in detail. These reports were the basis for ANL-W's summary report on the demonstration project (Benedict et al. 1999).

To assist in monitoring the progress of the demonstration project, DOE requested that the National Research Council establish a review committee, the Committee on Electrometallurgical Techniques for DOE Spent Fuel Treatment, to evaluate the technology and its development. Working with DOE and the National Research Council committee, ANL-W established four criteria for evaluating the demonstration. The evaluation criteria for the electrometallurgical spent fuel demonstration project are listed below (Benedict et al. 1999).

Criterion 1: *Demonstrate that 100 driver and up to 25 blanket EBR-II assemblies can be treated in a Fuel Conditioning Facility within three years, with a throughput rate of 16 kilograms per month for driver assemblies sustained for a minimum of three months, and a blanket spent nuclear fuel throughput rate of 150 kilograms per month sustained for one month.*

Criterion 2: *Quantification (for both composition and mass) of recycle, waste, and product streams that demonstrate projected material balance with no significant deviations.*

Criterion 3: Demonstrate an overall dependable and predictable process considering uptime, repair and maintenance, and operability of the linked process steps.

Criterion 4: Demonstrate that safety risks, environmental impacts, and nuclear materials accountancy are quantified and acceptable within regulatory limits.

Based on a comparison of the demonstration results with the above criteria for success, the demonstration project was a technical success. All key performance criteria were met or exceeded. The results of the demonstration project proved the technical feasibility of using electrometallurgical treatment technology to process DOE's inventory of sodium-bonded spent nuclear fuel. In addition, the demonstration project validated the throughput rate of the sodium-bonded spent nuclear fuel, quantified all process streams, fine-tuned the operational parameters, refined the electrometallurgical treatment equipment, and provided actual waste forms for characterization. This last accomplishment was of particular importance because, as the Defense Waste Vitrification Project at SRS has shown, waste characterization is a lengthy process. Waste forms must be subjected to detailed chemical analysis and long periods of exposure to expected repository conditions. The waste form characterization in the electrometallurgical treatment demonstration project has already initiated the waste acceptance process. Preliminary results of waste form testing indicate that both the metal and ceramic waste forms produced by the electrometallurgical process appear to be comparable to borosilicate glass, which has been tested and analyzed extensively under conditions relevant to a geologic repository.

The review committee of the National Academy of Sciences' National Research Council has continuously reviewed the progress of the Electrometallurgical Treatment Research and Demonstration Project and all reports to date have found the process to be proven for treating sodium-bonded spent nuclear fuel (Benedict et al. 1999).

In the most recent status report issued in the summer of 1999 (NAS 1999), the National Research Council Committee expressed some concerns about the long-term performance and potential releases from the waste forms under repository conditions. However, as noted above, work completed at ANL-W since the latest National Research Council review of the project indicates that both the ceramic and metallic electrometallurgical treatment waste forms would be comparable to borosilicate glass, which has been tested extensively under conditions relevant to the repository. The final report from the National Research Council Committee was published in April 2000. The National Research Council's final report on *Electrometallurgical Techniques for DOE Spent Fuel Treatment* concluded that "The EBR-II demonstration project has shown that the electrometallurgical technique can be used to treat sodium-bonded spent nuclear fuel." The report further stated that "The committee has found no significant technical barriers in the use of electrometallurgical technology to treat EBR-II spent fuel, and EMT [electrometallurgical treatment] therefore represents a potentially viable technology for DOE spent nuclear fuel treatment." DOE will consider the Council's final report during the Record of Decision process which follows the issuance of the final EIS.

1.7 CHANGES FROM THE DRAFT EIS

In response to comments on the SBSNF Draft EIS and as a result of information that was unavailable at the time of the draft EIS issuance, the final EIS contains revisions and new information. These revisions and new information are indicated by a double underline for minor word changes or by a sidebar in the margin for sentence or larger additions. Appendix A contains the comments received during the public review period of the SBSNF Draft EIS and DOE's responses to these comments. Responses to comments related to cost and nuclear nonproliferation issues, although included in the Appendix, did not result in any changes to the EIS. A brief discussion of the most important changes included in the final EIS is provided in the following paragraphs.

Results of the Electrometallurgical Treatment Research and Demonstration Project

As a result of public concern that results of the demonstration project were not incorporated in the draft EIS, a section (Section 1.6.3) was added in the final EIS with a description, status, and results of the demonstration project.

Justification of Purpose and Need and Timing

As a result of public concern that the draft EIS did not adequately justify the need and timing for the proposed action, Section 1.2 was revised to reflect DOE's position and DOE's responses to the related comments.

Relationship to Other NEPA Actions

The final EIS was revised to update the information provided on the *Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada,* which was issued in July 1999 (DOE 1999b).

Sodium Removal and Disposition

As a result of public comment, the description of an alternate method for decladding and cleaning sodium-bonded blanket spent nuclear fuel, the laser declad and alcohol wash process, was added in Section 2.3.9. The reason the process was not included in the evaluation of the reasonable alternatives also is included in Section 2.3.9.

No Action Alternative Definition

One of the two options of the No Action Alternative was revised from "indefinite" storage until the development of a currently less mature technology to "continued storage of the sodium-bonded spent nuclear fuel until 2035 or until the development of a currently less mature technology." The revision clarifies the issue raised by public comments concerning the time period covered by this EIS. This EIS covers the time period until 2035.

In addition, under both options of the No Action Alternative, it was determined that the sodium-bonded spent nuclear fuel would be packaged at ANL-W in preparation for shipment out of the State of Idaho by 2035.

No Action Alternative Assumptions

As a result of public comment, the assumption for the calculation of the radiological gaseous emissions under the No Action Alternative was changed. The draft EIS conservatively assumed that the radiological gaseous emissions would be a fraction of the total radiological gaseous emissions presented in the Programmatic Spent Nuclear Fuel EIS, in direct proportion to the heavy metal mass ratio of the sodium-bonded spent nuclear fuel to the total spent nuclear fuel stored at INEEL. The final EIS directly calculates the radiological gaseous emissions using a more realistic fuel degradation assumption based on historical evidence. This change considerably reduced the estimated radiological gaseous emissions as well as the resulting doses to workers and the public under the No Action Alternative.

Dose and Risk Calculations

As a result of public comments and the availability of recent data from the Electrometallurgical Treatment Research and Demonstration Project and the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000), dose calculations were revised in the final EIS. These

revisions include: (1) the addition of project total doses to workers, (2) the project total risk to the public under normal operations, and (3) changes in doses and risks to the public and workers from accidents. In addition, dose and risk values were rounded, resulting in some changes in the numerical values in the EIS.

Air Quality

Based on public comments on the draft EIS, concentrations and emissions from sources that operate in support of the processing alternatives at ANL-W (e.g., emergency generators) were quantified and added to Sections 3.2.3 and 3.3.3 (Air Quality and Noise) and Chapter 4 of the final EIS. In addition, the baseline nonradiological air quality concentrations for INEEL presented in the draft EIS were replaced with more current emission inventory data.

Land Use/Ecology

As a result of comments received on the draft EIS, reference to the newly established 29,950-hectare (74,000-acre) INEEL Sagebrush Steppe Ecosystem Reserve was added to Sections 3.2.1.1 (Land Use) and 3.2.6 (Ecological Resources) of the final EIS.

Water Quality

As a result of public comments, a discussion and a summary table of radioactive liquid effluent at both INEEL and SRS were added to Sections 3.2.4 and 3.3.4 (Water Resources) of the final EIS.

Geology and Soils

As a result of public comments on the draft EIS, material on earthquake activity and volcanism in the vicinity of INEEL (Section 3.2.5, Geology and Soils) was revised.

Existing Human Health Risk

As a result of public comments, baseline concentrations and associated hazard indexes or cancer risks for hazardous chemicals at both ANL-W and SRS were added to Sections 3.2.10 and 3.3.10 (Existing Human Health Risk) of the final EIS.

Waste Management

Records of Decision for the Final Waste Management Programmatic EIS (DOE 1997) addressing the management of high-level radioactive waste and low-level and mixed low-level radioactive waste were issued on August 26, 1999 (64 FR 46661), and February 25, 2000 (65 FR 10061), respectively. A summary of these decisions was added to the waste management discussion for both INEEL and SRS (Sections 3.2.11 and 3.3.11, respectively).

Cumulative Impacts

The cumulative impacts section (Section 4.11) was updated to reflect recent information obtained from the *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement* (DOE 1999d) and the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000).

Electric Energy Consumption

Section 4.14.3, Irreversible and Irretrievable Commitments of Resources, was revised to include electrical energy consumption associated with the alternatives under the proposed action.

Settlement Agreement and Consent Order with the State of Idaho

As a result of public comments concerning the contents of the Settlement Agreement and Consent Order with the State of Idaho, the entire document was included in the final EIS as Appendix K.

Melt and Dilute Driver Fuel at SRS

The option of using the melt and dilute process to treat sodium-bonded driver spent nuclear fuel at SRS was considered at the recommendation of a public comment. The option was dismissed from further evaluation as explained in the revised Section 2.6 of the final EIS.

Preferred Alternative

In accordance with requirements of the Council on Environmental Quality regulations (40 CFR 1502.14e), the final EIS incorporates DOE's Preferred Alternative for the treatment and management of sodium-bonded spent nuclear fuel. The Preferred Alternative is discussed in Section 2.8.

Transportation

The analysis was expanded to include the impacts from transporting the various waste forms and spent nuclear fuel packages from ANL-W to the INEEL Dry Storage Facility prior to transporting materials out of the State of Idaho by 2035.

Miscellaneous Revisions and Editorial Changes

Several sections in the SBSNF Final EIS were revised to reflect the availability of more recent data or to include corrections, improvements in the presentation, and other editorial changes. None of these revisions affects the environmental analysis presented in the EIS.

1.8 ORGANIZATION OF THE EIS

This EIS volume contains 9 chapters and 12 appendices. The main analyses are included in the chapters and additional project information is provided in the appendices. The 9 chapters provide the following information:

Chapter 1—Introduction: Background on the disposition of spent nuclear fuel; purpose and need for the proposed action; issues identified during the scoping and public comment periods; decisions to be made; and relationship of this EIS to other DOE NEPA actions and programs

Chapter 2—Proposed Action and Alternatives: Descriptions of sodium-bonded spent nuclear fuel; spent nuclear fuel treatment methods; spent nuclear fuel management facilities; alternatives considered; background information on the ultimate disposition of spent nuclear fuel; Preferred Alternative; and summary comparison of environmental impacts

Chapter 3—Affected Environment: Aspects of the environment that could be affected by the EIS alternatives

Chapter 4—Environmental Consequences: Analyses of the potential impacts of the EIS alternatives on the environment

Chapter 5—Environmental Laws, Regulations, and Consultations: Environmental, safety, and health regulations that would apply for this EIS's alternatives and the agencies consulted for their expertise

Chapters 6-9—Glossary; a list of preparers; a list of agencies, organizations, and persons to whom copies of this EIS were sent; and an index

The 12 appendices contain the following information: overview of the public participation process (scoping meetings and public comment period) and comment disposition; methods for assessing environmental impacts; detailed technology descriptions; characteristics of sodium-bonded spent nuclear fuel; normal operational impacts on human health; facility accident impacts on human health; evaluation of human health effects of overland transportation; environmental justice analysis; scientific terminology for ecological resources; *Federal Register* notices; the Settlement Agreement and Consent Order with the State of Idaho; and the contractor disclosure statement.

1.9 REFERENCES

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2. PROPOSED ACTION AND ALTERNATIVES

Chapter 2 describes the proposed action and alternatives to treat and manage the U.S. Department of Energy's inventory of sodium-bonded spent nuclear fuel and the No Action Alternative. The chapter includes discussions on the characteristics and quantities of the sodium-bonded spent nuclear fuel under consideration, the proposed treatment methods, and the potential sites and facilities for treatment or storage. It discusses why certain alternatives were dismissed from consideration. It also addresses issues associated with the ultimate disposition of the spent nuclear fuel and provides a summary comparison of the environmental impacts associated with the proposed action and the No Action Alternative.

2.1 INTRODUCTION

To fulfill the purpose and need discussed in Section 1.2, the U.S. Department of Energy (DOE) proposes to treat and manage sodium-bonded spent nuclear fuel and facilitate ultimate disposal in a geologic repository. The reasonable alternatives are determined by the technology options available to DOE to treat and manage the sodium-bonded spent nuclear fuel. To assist the reader in understanding the proposed action and reasonable alternatives, the following sections provide background information on the characteristics, inventory, and current storage locations of the sodium-bonded spent nuclear fuel; a discussion on the Electrometallurgical Treatment Research and Demonstration Project; the technology alternatives for its treatment and management; and the locations where these technologies could be implemented. The reasonable alternatives are discussed in Section 2.5.

2.2 SODIUM-BONDED SPENT NUCLEAR FUEL

As a result of research, development, and demonstration activities associated with liquid metal fast breeder reactors, DOE has approximately 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel in its inventory. This represents approximately 2 percent of DOE's total current spent nuclear fuel inventory of nearly 2,500 metric tons of heavy metal. The common characteristic of sodium-bonded spent nuclear fuel is the presence of metallic sodium in the space between the cladding and the fuel and/or within the mass of the fuel. The presence of this chemically reactive material necessitates DOE's consideration of suitable treatment and management alternatives for this spent nuclear fuel before disposal in a geologic repository. Detailed descriptions of the characteristics of the sodium-bonded spent nuclear fuel in DOE's inventory are included in Appendix D.

Proposed Action

DOE proposes to treat and manage the sodium-bonded spent nuclear fuel in a safe and efficient manner and facilitate disposal in a geologic repository.

The bulk of sodium-bonded spent nuclear fuel in DOE's inventory is of two general types: driver fuel and blanket fuel. Driver fuel is used mainly in the center of the reactor core to "drive" and sustain the fission chain reaction. It is highly enriched in the fissile isotope uranium-235. Blanket fuel is made from depleted uranium, a type of uranium in which most of the fissile uranium-235 has been removed. Blanket fuel usually is placed at the perimeter of the core and is used to breed the fissile material plutonium-239. It contains primarily the nonfissile isotope uranium-238, which converts to fissile plutonium-239 with the absorption of neutrons produced from the fission process. In some cases, as in the case of the Experimental Breeder Reactor-II (EBR-II), blanket fuel also has been used at the perimeter of the core for shielding. Typically, the fuel matrix in the sodium-bonded spent nuclear fuel is a uranium alloy or

uranium metal. A very small quantity (approximately 0.1 percent in mass of heavy metal) is in the form of uranium oxide, uranium or plutonium nitride, and uranium or plutonium carbide. Typical driver and blanket spent nuclear fuel elements are shown schematically in **Figure 2–1**.

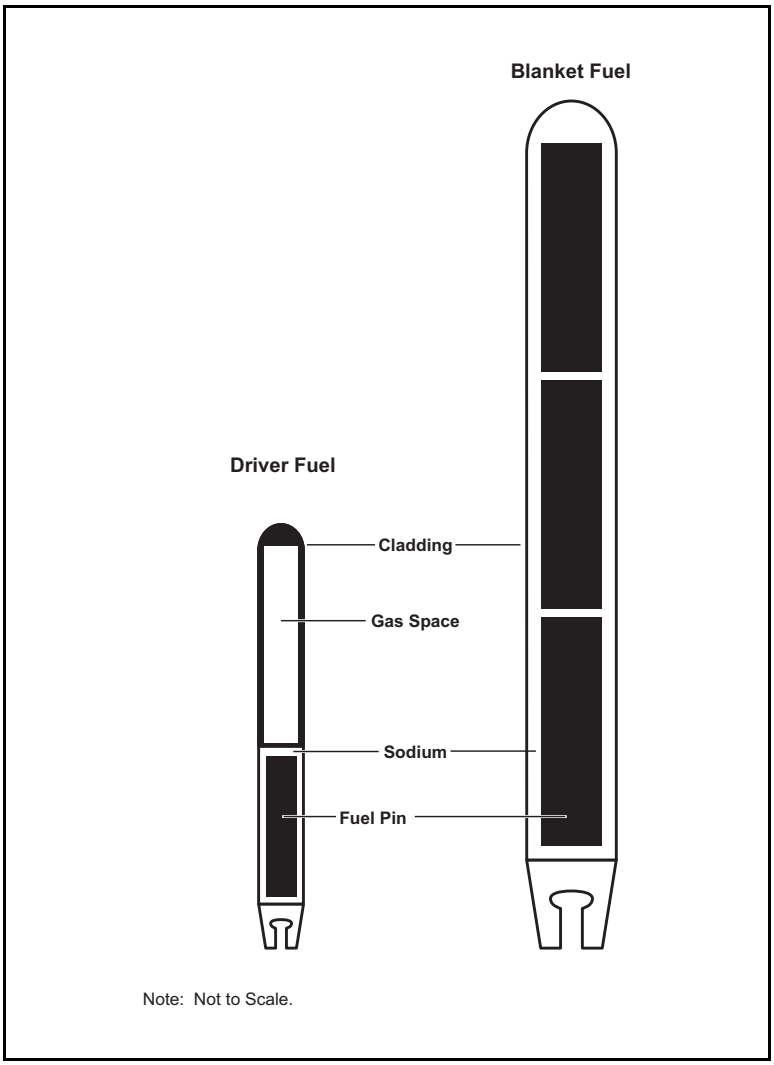

Figure 2–1 Typical Driver and Blanket Spent Nuclear Fuel Elements

The blanket and driver spent nuclear fuel addressed in this environmental impact statement (EIS) contain metallic sodium between the cladding and the metallic fuel pins to improve the heat transfer from the fuel to the reactor coolant through the stainless steel cladding. When driver fuel is irradiated in the reactor for some period of time, the metallic fuel swells as fission products are generated until it reaches the cladding wall. Pores form throughout the fuel as it swells under pressure from the gaseous fission products. As these pores expand and connect to one another, the fission gases escape to a plenum in the fuel element just above the metallic fuel. As the gas escapes, the liquid sodium flows into these tiny pores, much like a sponge. As more pores form and grow, others are closed off from the fuel surface, including those containing sodium. Between 20 and 40 percent of the available sodium (up to 0.8 grams) may enter the driver fuel and become inseparable from the uranium except by dissolving or melting the fuel (Hofman and Walters 1994).

It also is well documented (Hofman and Walters 1994) that fuel and cladding components interdiffuse during irradiation to such an extent that mechanical stripping of the driver fuel cladding is not practical. The gap between the cladding and the fuel pin that contains sodium early in the irradiation lifetime disappears gradually due to outward swelling of the fuel pin. After approximately a 1 to 3 percent burnup, this gap is closed by swelling of the fuel pin due to irradiation effects such as fission gas bubble growth. Once contact between the fuel pin outer surface and cladding inner surface is made, cladding constituents (mainly nickel, chromium, and iron) gradually interdiffuse with fuel constituents (mainly uranium, plutonium, and zirconium) and the rare earth fission products (neodymium, cerium, lanthanum, praseodymium, samarium, and promethium) in the fuel. A solid-state layer bonding the fuel and cladding together is formed. This interdiffused layer effectively attaches the cladding to the fuel pin permanently in localized regions. Mechanical forces applied to these regions in a decladding operation either would leave pieces of the fuel pin attached to the cladding or vice versa. The resulting mix of cladding and fuel still would contain sodium inaccessible to subsequent treatment.

Blanket fuel, on the other hand, is at such a low burnup that significant swelling of the fuel pin does not occur. Gaps between the fuel pin and the cladding still exist at low burnup and little or no interdiffusion takes place. Cladding therefore can be mechanically removed from blanket fuel. Also, the swelling of this fuel is still low enough at typical blanket burnup that very little interconnected porosity exists. Hence, minimal sodium trapping would take place. Therefore, blanket fuel could be declad to effect sodium removal. The sodium removed from the sodium-bonded blanket spent nuclear fuel could be distilled and stabilized for disposal as low-level radioactive waste. The cladding would be disposed of as low-level radioactive waste or as part of the waste form being qualified for the proposed geologic repository.

2.2.1 EBR-II Spent Nuclear Fuel

EBR-II was a research and test reactor at Argonne National Laboratory-West (ANL-W) used to demonstrate the engineering feasibility of a sodium-cooled, liquid metal reactor with a steam electric power plant and integral fuel cycle. It achieved initial criticality in September 1961 and continued to operate until September 1994. During the 33 years of operation, numerous fuel designs were tested in EBR-II. EBR-II spent nuclear fuel contains both driver and blanket fuel.

The EBR-II driver spent nuclear fuel is stainless steel-clad, highly enriched uranium in a uranium alloy, typically either zirconium or fissium (an alloy of molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium). There are some variations in the specific cladding alloys, enrichments, fuel compound alloys, dimensions, and burnup. When the fuel is "spent," the enrichment (ratio of uranium-235 to total uranium) ranges between 55 and 76 percent. Each driver spent nuclear fuel element has a metal fuel pin about 36 centimeters (14 inches) long and less than 0.5 centimeters (0.2 inches) in diameter. The typical EBR-II driver spent nuclear fuel pin is a metal alloy of either 90 percent uranium and 10 percent zirconium or 95 percent uranium and 5 percent fissium. This fuel pin and a small amount of metallic sodium are loaded into a 74-centimeter-long (29-inch-long) stainless steel tube (cladding) and welded shut. This unit of fuel is called an element. Sixty-one (91 for some fuel) fuel elements are put together in a stainless steel hexagonal duct to make a fuel assembly approximately 2.3 meters (92 inches) long and 5.8 centimeters (2.3 inches) across. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Appendix D.

The EBR-II blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in metallic form. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary differences between the blanket designs are the dimensions. In EBR-II, the blanket assemblies were used primarily for shielding and for reducing the required size of the reactor core. Blanket assemblies were placed outside of a stainless steel shield for all but the first few years of EBR-II operation. Blanket assemblies are similar to driver assemblies, except that the individual blanket pins are larger. The blanket pins, made entirely from depleted uranium, are 1.1 centimeters (0.4 inches) in diameter. Three to five pins placed end-to-end make a sodium-bonded blanket element between 84 and 140 centimeters (33 to 55 inches) long. Since the blanket pins have a larger diameter, 19 blanket elements comprise a blanket assembly. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Appendix D.

The fuel from the last seven years of EBR-II operation is presently stored in three different locations at ANL-W (the Fuel Conditioning Facility, the Hot Fuel Examination Facility, and the Radioactive Scrap and Waste Facility) and two different locations at the Idaho Nuclear Technology and Engineering Center (INTEC), formerly the Idaho Chemical Processing Plant (CPP). Previously, spent nuclear fuel was shipped to INTEC for reprocessing. However, INTEC ceased accepting the fuel in 1991 when a new uranium-zirconium alloy fuel, which could not be dissolved with INTEC's existing plutonium-uranium extraction (PUREX) reprocessing system, went into full use at EBR-II. Prior to that, approximately 6 metric tons of EBR-II fuel were processed at INTEC. When DOE stopped processing at INTEC in 1992, elements from some 500 EBR-II driver spent

nuclear fuel assemblies of earlier design were left in storage pools (CPP-603 and CPP-666) located at INTEC. Water has been observed leaking into some of the storage containers in the CPP-603 storage pool, and the EBR-II fuel inside has reacted with the water and produced hydrogen gas. This is one of the reasons DOE is planning to remove all the spent nuclear fuel from the CPP-603 storage pool and place it in dry storage. National Environmental Policy Act (NEPA) coverage for this activity is provided by the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (Programmatic Spent Nuclear Fuel EIS) (DOE 1995a), and is not within the scope of this EIS.

2.2.2 Fermi-1 Spent Nuclear Fuel

Fermi-1 was designed and built at Monroe, Michigan (30 miles southwest of Detroit), to demonstrate the feasibility of the sodium-cooled, liquid metal fast breeder reactor for electric power production. Fermi-1 was a sodium-cooled, fast reactor. The reactor achieved initial criticality in 1963 and operated until September 1972. Fermi-1 was licensed for operation at a power level of 200 megawatts-thermal. Only blanket fuel from Fermi-1 is sodium-bonded.

The Fermi-1 blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in a uranium-molybdenum alloy. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary differences between these blanket designs are dimensions, elements per assembly, and burnup. Fermi-1 blanket elements are similar to EBR-II blanket elements in enrichment, but differ in dimensions (Fermi-1 elements are larger), form (uranium-molybdenum alloy versus uranium metal), and burnup. Because of its lower burnup, the Fermi-1 blanket fuel, which contains only about 0.2 percent plutonium by weight compared to approximately 1 percent plutonium by weight for the EBR-II blanket fuel, is subject to less stringent safeguard and security requirements than the EBR-II blanket fuel. This is an important consideration in the cost of managing the storage of these two types of fuel.

After the Fermi-1 reactor was permanently shut down, the blanket assemblies were placed into 14 canisters and transported to INTEC in 1974 and 1975 in 14 shipments. The canisters are made of stainless steel with a carbon steel basket inside. The canisters are 3.46 meters (134 inches) long and 64.8 centimeters (25.5 inches) in diameter. Twelve of the canisters contain the radial blanket subassemblies and 2 of the canisters contain the shorter axial blanket subassemblies. A subassembly is a cut assembly containing the blanket fuel pins. The canisters were placed into CPP-749, which is an underground dry storage system. The 14 canisters are stored in a single row of vaults on 4.6-meter (15-foot) centers.

The total quantity of Fermi-1 blanket material, both axial and radial, is 34 metric tons of heavy metal. The blanket assemblies have a very low irradiation history. Therefore, the inventory of fission products, activation products, and transuranics is low. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Appendix D.

2.2.3 Fast Flux Test Facility and Miscellaneous Sodium-Bonded Spent Nuclear Fuel

DOE's inventory of sodium-bonded spent nuclear fuel includes eight liquid metal reactor test assemblies containing driver spent nuclear fuel that were irradiated at the Fast Flux Test Facility at Hanford, Washington. It also includes small quantities of fuel that have metallic sodium or the alloy of sodium and potassium from liquid metal reactor experiments. These miscellaneous small-lot fuels differ in cladding composition, uranium content, enrichment, and burnup. Some of the fuel consists of uranium and/or plutonium carbides, nitrides, and oxides in addition to metal uranium or alloy. For the purposes of this EIS, this miscellaneous fuel is assumed to have driver fuel characteristics. This fuel is located at several DOE sites such as the Hanford site, Oak Ridge National Laboratory, the Savannah River Site (SRS), Sandia National Laboratories/New Mexico, and the Idaho National

Engineering and Environmental Laboratory (INEEL). Those lots stored outside INEEL will be transported to INEEL pursuant to the amended Record of Decision (61 FR 9441) for the Programmatic Spent Nuclear Fuel EIS (DOE 1995a). Under the proposed action, it is assumed that this spent fuel will be stored at INEEL.

Table 2–1 provides a summary of the spent nuclear fuel addressed by this EIS. As described earlier, the majority of the spent nuclear fuel consists of EBR-II driver fuel, EBR-II blanket fuel, and Fermi-1 blanket fuel.

Table 2–1 Overview of Sodium-Bonded Spent Nuclear Fuel Categories

Spent Nuclear Fuel Type	*Storage Volume* [a] *(cubic meters)*	*Metric Tons of Heavy Metal*	*Sodium Content (kilograms)*
EBR-II driver	58 [b]	3	83
EBR-II blanket	13	22	173
Fermi-1 blanket	19	34	365
Fast Flux Test Facility driver	8 [b]	0.3	7
Miscellaneous [c]	3 [b]	0.1	31
Total	101	60	662

[a] Volume refers to the canister storage volume.
[b] A larger volume per unit mass for the driver spent nuclear fuel is required for criticality control.
[c] Assumed to have driver fuel characteristics.

Table 2–2 provides the site where the sodium-bonded spent nuclear fuel is stored, the locations within the DOE site, and the various storage configurations within the storage site.

Table 2–2 Sodium-Bonded Spent Nuclear Fuel Storage Locations and Configurations

Spent Nuclear Fuel Type	*Current Storage Locations and Configurations*		
	DOE Site	*Location*	*Configuration*
EBR-II driver	INEEL (ANL-W)	Radioactive Scrap and Waste Facility	Loose elements in canisters
		Hot Fuel Examination Facility	Loose elements
		Fuel Conditioning Facility	In process material[a]
EBR-II blanket	INEEL (ANL-W)	Radioactive Scrap and Waste Facility	Elements in canisters
		Fuel Conditioning Facility	In process material[a]
EBR-II driver	INEEL (INTEC)	CPP-603 basin	About 12 elements per canister
		CPP-666 basin	
Fermi-1 blanket	INEEL (INTEC)	CPP-749 dry well underground	Cut/uncut assemblies in 14 storage canisters
Fast Flux Test Facility driver	INEEL (ANL-W)	Hot Fuel Examination Facility	Loose elements
	Hanford	Fast Flux Test Facility, Buildings 405 and 403	Intact assemblies
Miscellaneous	Sandia National Laboratories/ New Mexico	Technical Area V	Experimental capsule
	SRS	Receiving Basin for Offsite Fuel	Elements
	Oak Ridge National Laboratory	Building 3525	Elements

[a] Processed as part of the EBR-II Electrometallurgical Treatment Research and Demonstration Project.

2.3 TREATMENT AND MANAGEMENT METHODS

DOE has identified several potential treatment, management, and packaging methods that could be used to prepare sodium-bonded spent nuclear fuel for disposal in a geologic repository. These are: the electrometallurgical process; the PUREX process; packaging in high-integrity cans; the melt and dilute process; the glass material oxidation and dissolution system (GMODS) process; the direct plasma arc-vitreous ceramic process; and the chloride volatility process. Each of these methods is discussed below. In formulating reasonable alternatives under the proposed action, the GMODS process, the direct plasma arc process, and the chloride volatility process were not considered sufficiently mature technologies to be included as reasonable alternatives (see Section 2.6).

Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository without treatment, i.e., packaging the fuel in high-integrity cans without sodium removal, has been considered in this EIS under the No Action Alternative. The option may not meet current U.S. Nuclear Regulatory Commission (NRC) and/or Resource Conservation and Recovery Act (RCRA) requirements.

2.3.1 Electrometallurgical Treatment Process

The electrometallurgical treatment process was developed at the Argonne National Laboratory for processing EBR-II driver and blanket spent nuclear fuel assemblies with metallic fuel. The process has been demonstrated for the stainless steel-clad uranium alloy fuel used in that reactor. Modifications to the process are used for the treatment of oxide, nitride, and carbide sodium-bonded spent nuclear fuel. The electrometallurgical treatment process uses electrorefining, which is an industrial technology used to produce pure metals from impure metal feedstock (DOE 1996). Electrorefining has been used to purify metal for more than 100 years. **Figure 2–2** illustrates the various steps within the electrometallurgical treatment process at ANL-W.

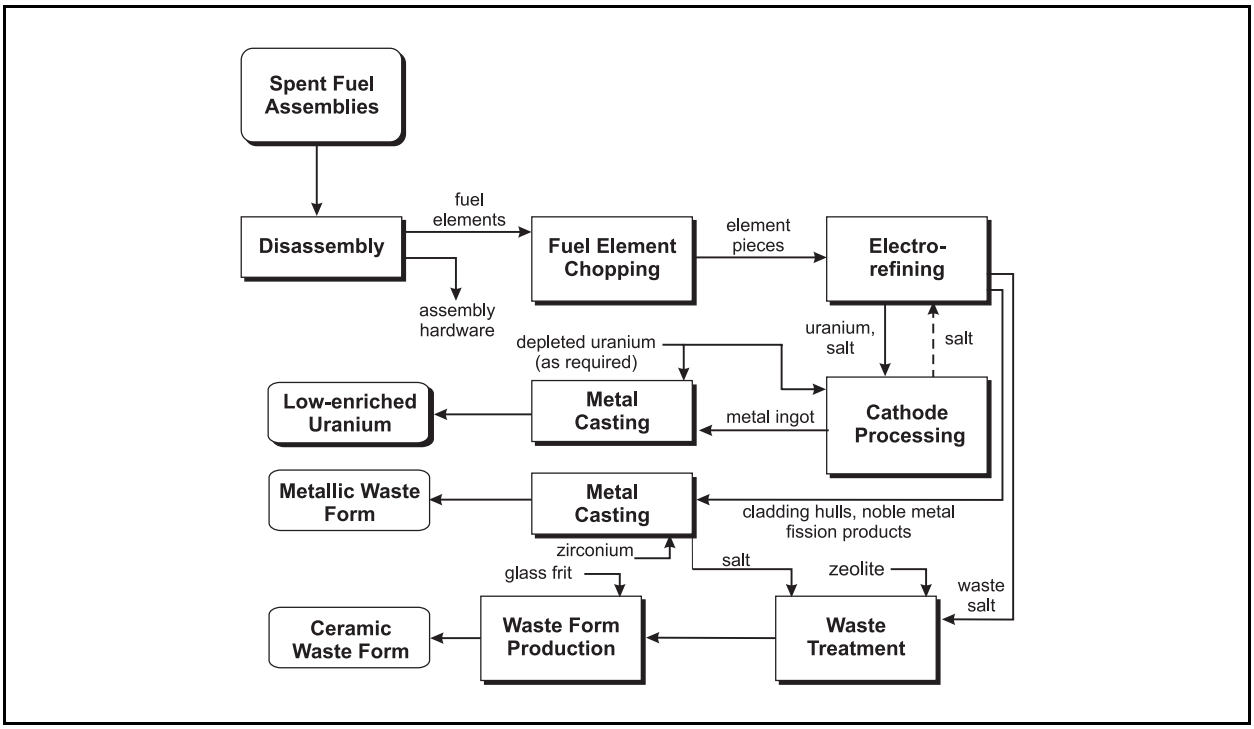

Figure 2–2 Electrometallurgical Treatment Process Flow Diagram

The first step in processing sodium-bonded spent nuclear metallic fuel involves the removal of fuel elements from the fuel assemblies. The fuel elements then would be chopped into short segments and placed in stainless steel baskets to form the anode in the electrorefiner.

The electrorefiner, in which the electrometallurgical treatment occurs, would be maintained at about 500 °C (930 °F) and contain a molten mixture of primarily two salts, lithium chloride and potassium chloride. The chopped fuel elements in the anode baskets would be lowered into the molten salt. Upon application of an electric voltage between the anodes and cathodes, uranium, transuranic elements including plutonium, most of the fission products, and the sodium would dissolve into the salt. The uranium would be deposited by the current at the cathode. The stainless steel cladding hulls and some of the insoluble fission products (i.e., noble metals) would remain in the anode baskets.

After a sufficient amount of spent nuclear fuel has been treated, the salt would be removed and solidified. The salt, which contains the sodium in the form of sodium chloride, transuranic elements, and most fission products extracted from the spent nuclear fuel, would be solidified, ground to a desired size, and mixed with zeolite. Zeolite is any of a group of alumina silicate minerals used as filters and ion-exchange agents. Zeolite is used to collect certain fission products from the process salt. The fission products, sodium, and transuranics, including plutonium in the salt and zeolite, would be heated so the salt becomes sorbed into the zeolite structure. Glass frit then would be added to the zeolite mixture and hot-pressed to produce a ceramic high-level radioactive waste form that is expected to be suitable for disposal.

The uranium deposited at the cathode would be removed from the electrorefiner and treated to remove any adhered salts. Then the uranium would be melted (and depleted uranium added if necessary), solidified to form an ingot, and further processed in a metal casting furnace to produce low-enriched uranium ingots. The stainless steel cladding hulls and the insoluble fission products would be melted in the casting furnace to produce a metal high-level radioactive waste form that is expected to be suitable for disposal in a geologic repository.

The oxide fuel would be prepared for treatment using the electrometallurgical treatment process by reducing the uranium oxide to uranium metal with lithium metal dissolved in small batches of lithium chloride-potassium chloride molten salt solution. The resulting uranium-bearing solution would be added to the molten salt solution used in the electrometallurgical treatment process for other sodium-bonded fuel and blanket elements and would be processed with those materials.

The carbide fuel would be prepared for electrometallurgical treatment by cleaning the fuel of sodium to the extent possible and then converting the fuel to uranium oxide with water or diluted acid. This oxide then would be converted to uranium metal by lithium metal in a molten salt solution and processed by electrometallurgical treatment with other sodium-bonded spent nuclear fuel and blanket elements. The nitride fuel also would be prepared for electrometallurgical treatment by converting it to uranium fuel.

In addition to the metal and ceramic waste form, some low-level radioactive waste also would be generated during the disassembly process of the spent nuclear fuel assemblies in the form of hardware. A detailed description of the electrometallurgical treatment process is presented in Appendix C.

2.3.2 PUREX Process

The PUREX process is a counter-current solvent extraction method which has been used extensively throughout the world since 1954 to separate and purify uranium and plutonium from fission products contained in aluminum-clad spent nuclear fuel and irradiated uranium targets. PUREX is not a thermal process; therefore, it takes place at low temperatures. DOE has two operating facilities at SRS, F-Canyon

and H-Canyon, that use the PUREX process for treatment of aluminum-clad fuel and targets. Use of the PUREX process facilities at SRS for treating sodium-bonded spent nuclear fuel involves certain restrictions inherent in the design: 1) the presence of sodium complicates the process as employed in the front-end of the SRS facilities; 2) the presence of stainless steel cladding would require significant modifications or additions to the existing front-end of the facilities; and (3) the presence of alloys (e.g., zirconium) in some of the fuel is incompatible with the SRS dissolution process. For this reason, treatment of driver sodium-bonded spent nuclear fuel is not feasible without significant modification to the existing PUREX process. However, the SRS facilities could be used without modification for the blanket sodium-bonded spent nuclear fuel if it is declad and its sodium removed prior to the process. In such a case, the F-Canyon facility would be used.

The fuel pins would be dissolved in an aqueous solution of nitric acid. The resulting nitric acid solution containing uranium, plutonium, and fission products would undergo feed clarification (to remove settlable solids) and acidity/alkalinity adjustment. The clarified aqueous solution then would be treated via the PUREX process utilizing centrifugal contactors and separators that involve organic solvent washing to produce: (1) an aqueous high-level radioactive waste containing the bulk of the fission products, americium, and neptunium; (2) a material stream containing the recovered plutonium; and (3) a material stream containing the recovered uranium. The plutonium- and uranium-containing streams each would undergo a second cycle of solvent washing to further separate the residual fission products and actinides from the plutonium and uranium. The aqueous high-level radioactive waste eventually would be processed to a borosilicate glass form. Material streams from the PUREX process would be uranium oxide, plutonium metal, and high-level radioactive waste. The uranium oxide would be stored on site as depleted uranium. The plutonium, approximately 260 kilograms (572 pounds), would be disposed of in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999c). **Figure 2–3** illustrates the various steps necessary for the treatment of sodium-bonded spent nuclear fuel in conjunction with the PUREX process. A detailed description of the process is presented in Appendix C.

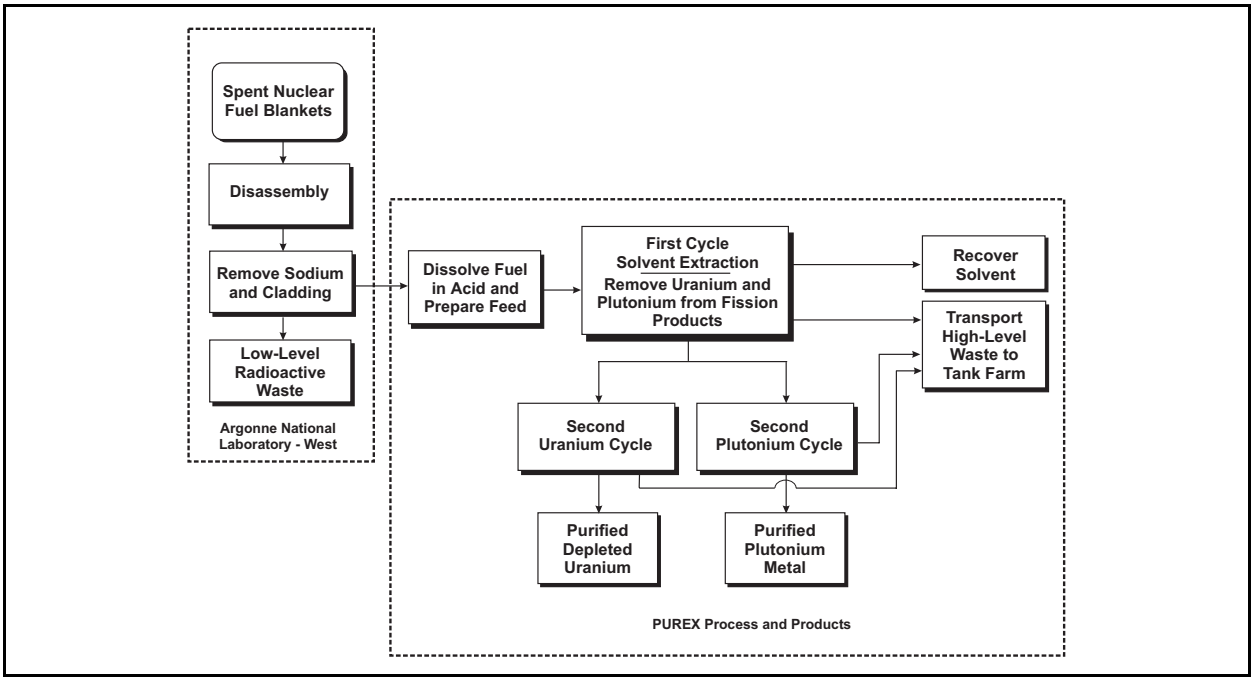

Figure 2–3 PUREX Process Flow Diagram

2.3.3 High-Integrity Can Packaging

The high-integrity can packaging provides substitute cladding for damaged or declad fuel, or another level of containment for intact fuel. The can would be used to store fuel on site until it is ready to be shipped to the repository. The can is constructed of a highly corrosion-resistant material (Hastelloy Alloy C-22) to provide corrosion protection during storage at the site. It also could provide long-term protection in a repository environment (i.e., for 1,000 or more years after repository closure with no institutional control). The high-integrity cans would be placed into a standardized canister for transportation and eventual placement in the repository in waste packages.

The analysis for packaging sodium-bonded spent nuclear fuel in high-integrity cans was performed with and without decladding and/or sodium removal. Packaging sodium-bonded blanket spent nuclear fuel in high-integrity cans with sodium removal was analyzed in the EIS under Alternative 2. Packaging sodium-bonded spent nuclear fuel in high-integrity cans without sodium removal was analyzed in this EIS as a direct disposal option under the No Action Alternative (see Sections 2.3.8 and 2.5.1).

The high-integrity cans would be placed in dry storage at ANL-W. Prior to emplacement in a repository, the high-integrity cans would be placed into a standardized canister designed to promote containment under repository conditions. **Figure 2–4** illustrates the high-integrity can flow process. A detailed description of the high-integrity can packaging is presented in Appendix C.

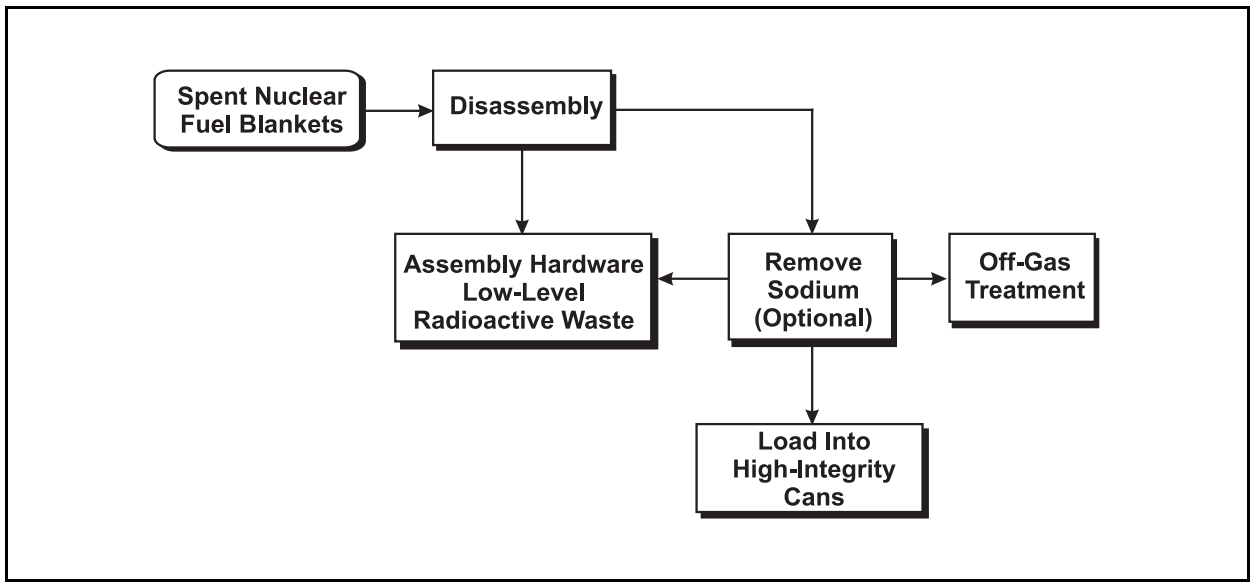

Figure 2–4 High-Integrity Can Packaging Flow Diagram

2.3.4 Melt and Dilute Process

There are three options for the melt and dilute process. In the first option, bare uranium blanket spent nuclear fuel pins with the sodium removed would be melted with aluminum at SRS using technology similar to that proposed for the aluminum-clad research reactor fuel. The second and third options would be conducted at the ANL-W site using metallurgical technology developed for uranium and stainless steel cladding. In the second option, blanket spent nuclear fuel elements would be melted with additional stainless steel. In the first two options, there would be no actual dilution of the fissile component of the uranium since it is present at only 0.2 percent, i.e., far less than the 0.7 percent in natural uranium. **Figure 2-5** illustrates the first two options of the melt and dilute process. The third option would involve a modified melt and dilute process

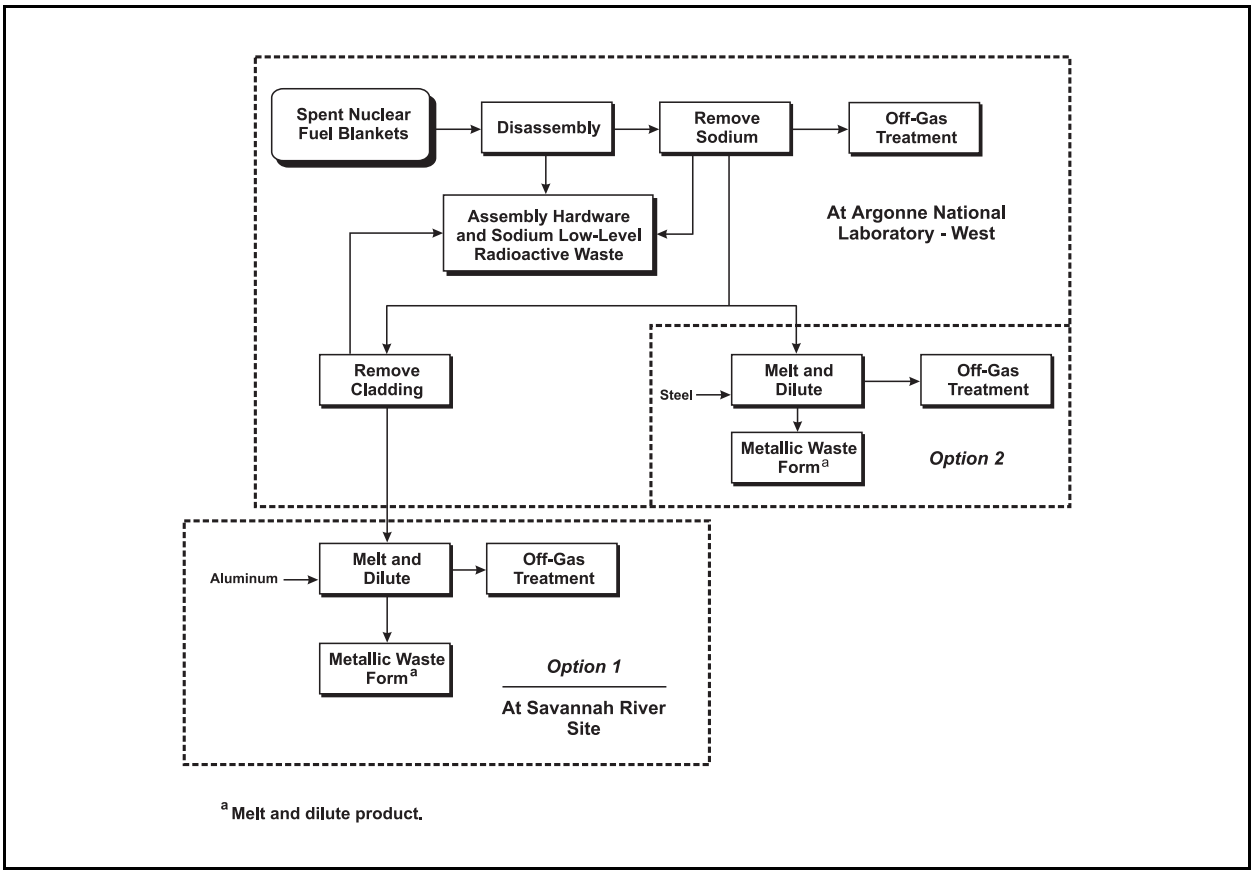

Figure 2–5 Melt and Dilute Process Flow Diagram (Options 1 and 2)

that would be capable of handling the sodium volatilized from processing chopped driver spent nuclear fuel elements with the cladding intact. **Figure 2-6** illustrates the third option of the melt and dilute process. A detailed description of the melt and dilute process is presented in Appendix C.

Option 1: Uranium-Aluminum Option for Blanket Pins

Declad and cleaned blanket pins would be received at SRS in aluminum canisters, each containing some 60 kilograms (132 pounds) of material. The canisters would be stored until they fit into the processing schedule. Following validation of the contents, the canisters would be loaded into a melting furnace with additional aluminum. The furnace would operate at a temperature of approximately 1,000 °C (1,830 °F), significantly in excess of the aluminum-uranium alloy melting temperature, to initiate melting within a reasonable time frame. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. A metal alloy ingot would be cast, sampled, and packaged.

Option 2: Uranium-Steel Option for Blanket Pins

Blanket elements with the sodium removed but not declad would be loaded into a furnace crucible. A small amount of radioactive waste steel might be added to the crucible to reach the desired uranium-steel composition. The furnace would be heated to approximately 1,400 °C (2,550 °F) to melt the uranium, after which the steel would be dissolved slowly into the uranium pool. The mixture would be stirred electromagnetically to a uniform composition. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. A metal alloy ingot would be cast, sampled, and packaged.

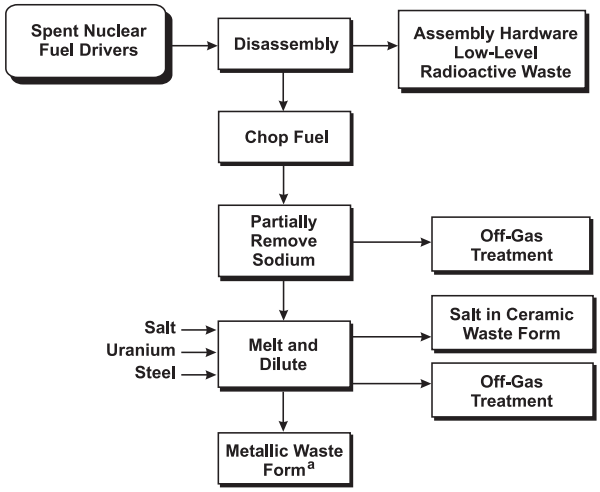

Figure 2–6 Melt and Dilute Process Flow Diagram (Option 3)

Option 3: Uranium-Steel Option for Driver Spent Nuclear Fuel

Some of the sodium in driver spent nuclear fuel elements would be removed in a similar manner to the sodium from blanket spent nuclear fuel elements. A modified melt and dilute process would be used for driver spent nuclear fuel still containing the cladding and some metallic sodium. The addition of flux salt is the only modification to the process required to capture residual sodium from the driver spent nuclear fuel. Chopped driver spent nuclear fuel elements would be loaded into an induction furnace and covered with a layer of low melting-temperature salt containing uranium, iron, or manganese chloride as a component to oxidize the molten sodium. Depleted uranium would be added in the ratio of about 2.5 to 1 to reduce the enrichment to less than 20 percent uranium-235. Radioactive waste steel would be added to complete the mixture. The use of radioactive waste steel reduces the inventory of the low-level radioactive waste. This furnace is operated at the same temperature as in Option 2. The sodium would react with and be captured in the flux salt, protecting the off-gas treatment filter banks. After the melt is mixed, a vacuum would be applied to complete the volatilization of the salt, which would be condensed and partially reused. The salt, which includes sodium in a nonreactive form, would be stabilized in a ceramic waste form similar to the waste form from the electrometallurgical treatment process. The metal melt would be stirred to achieve uniform composition and cast into an ingot, placed in a standardized canister, and stored.

The process described above can be used for sodium-bonded spent nuclear metallic fuel. Oxide, carbide, and uranium nitride fuel types cannot be treated using the melt and dilute process because of their high melting points.

2.3.5 GMODS Process

The GMODS process uses oxides to convert unprocessed spent nuclear fuel directly to borosilicate glass. The basic concept is to combine unprocessed sodium-bonded spent nuclear fuel and lead borate glass in a glass melter at a temperature of 800 to 1,000 °C (1,470 to 1,830 °F). The uranium and plutonium in the spent nuclear fuel would be converted into oxides and dissolved in the glass. Due to the powerful dissolution and oxidation properties of the lead borate glass melt, containment is a concern, and a water-cooled, cold-wall, induction-heated melter must be used. The waste form is borosilicate glass and would contain uranium,

the transuranic elements, the fission products, and the sodium present in the sodium-bonded spent nuclear fuel. As with all processes that dissolve or melt spent nuclear fuel, the GMODS process would produce radioactive off-gases. These gases would be filtered and treated by appropriate means. **Figure 2–7** illustrates the GMODS treatment process. A description of this process is presented in Appendix C.

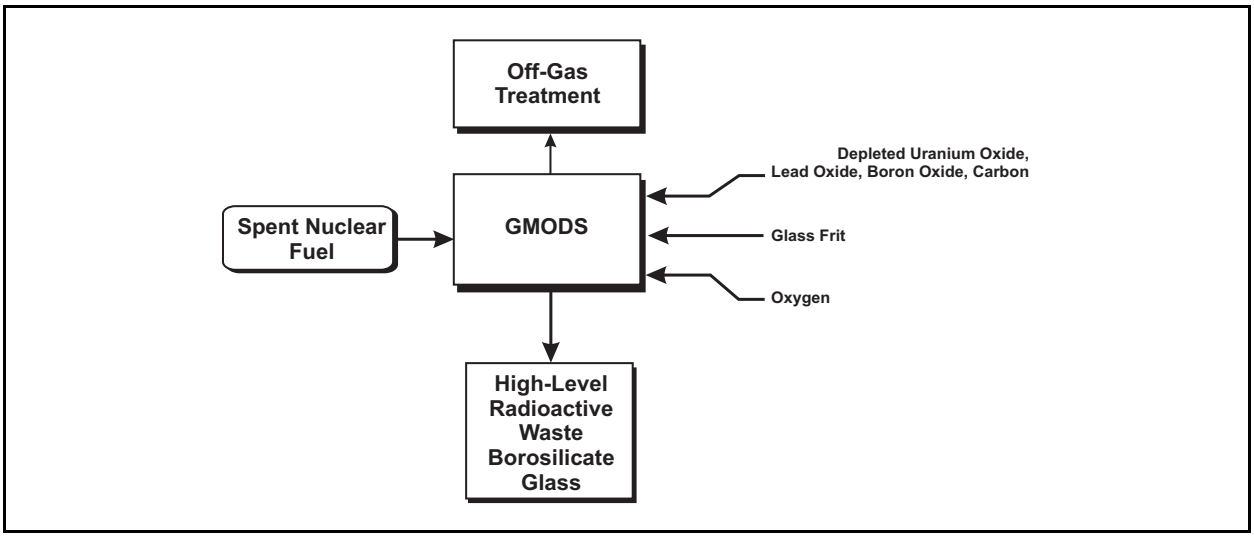

Figure 2–7 GMODS Process Flow Diagram

2.3.6 Direct Plasma Arc-Vitreous Ceramic Process

In this process, the sodium-bonded spent nuclear fuel would be cut into small pieces and melted and oxidized in a rotating furnace containing molten ceramic materials at a temperature of about 1,600 °C (2,900 °F) or higher. A direct-current plasma torch would supply the energy required in the process. Rotation would be used to keep the molten pool in the furnace. The spent nuclear fuel would be fed into the process with minimal pretreatment. Ceramic material would be added as necessary with the mixture homogenized by the torch. When the spent nuclear fuel is homogeneously melted and oxidized throughout the ceramic, rotation would be slowed to allow the molten vitreous ceramic to pour out by gravity flow into canister molds.

Metallic fuel such as the EBR-II spent nuclear fuel would require the addition of some ceramic material. Depleted uranium could be added to the process in almost any form to reduce the uranium-235 enrichment if necessary. Criticality issues would be addressed by limiting the process to batch runs of preselected quantities of fissile material with the addition of the depleted uranium and neutron poisons, if necessary.

As with all processes that dissolve or melt spent nuclear fuel, the plasma arc treatment would produce radioactive off-gases. These gases would be filtered and treated by appropriate means. **Figure 2–8** illustrates the direct plasma arc-vitreous ceramic treatment process. A description of this process is presented in Appendix C.

2.3.7 Chloride Volatility Process

The chloride volatility process is an advanced treatment technology that was investigated at INEEL. The process uses the differences in volatilities of chloride compounds to segregate major nonradiological constituents from spent nuclear fuel for the purpose of volume reduction, and isolates the fissile material to produce a glass or ceramic waste form. The major steps are: (1) high-temperature chlorination at about 1,500 °C (2,730 °F) and conversion of metallic fuel and cladding to gaseous chloride compounds; (2)

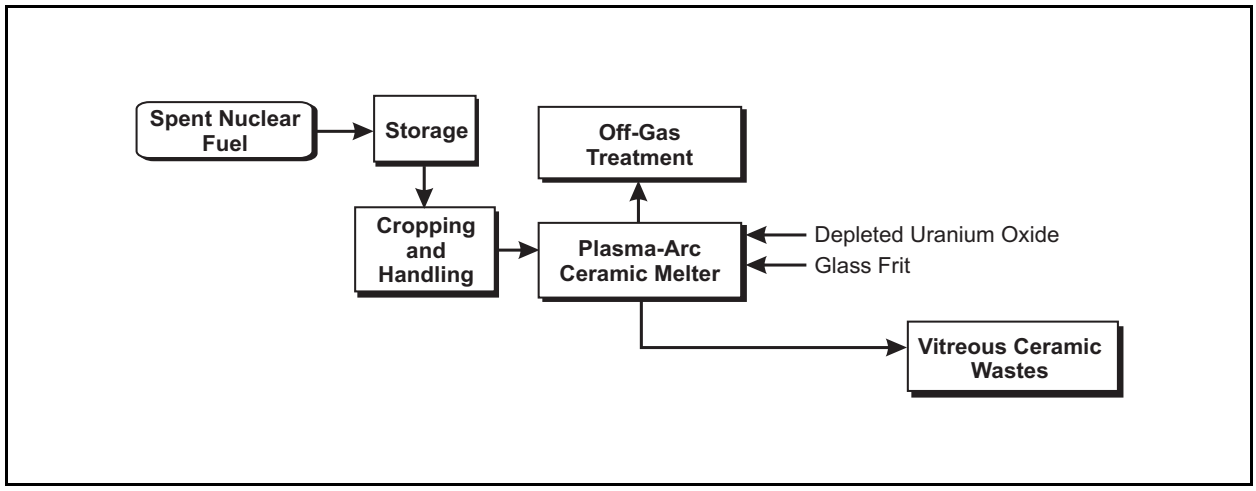

Figure 2–8 Direct Plasma Arc-Vitreous Ceramic Treatment Process Flow Diagram

removal of the transuranic chlorides and most of the fission products in a molten zinc chloride bed at approximately 400 °C (750 °F); (3) condensation of the other chlorides (e.g., uranium hexachloride) in a series of fluidized beds and condensers at successively lower temperatures; and (4) zinc chloride regeneration/recycling. The transuranics and fission product chlorides then would be converted into either fluorides or oxides for disposal. This process inherently handles volatilized fission products and chlorine gas, which presents significant unique occupational and public risks. **Figure 2–9** illustrates the chloride volatility treatment process. A description of this process is presented in Appendix C.

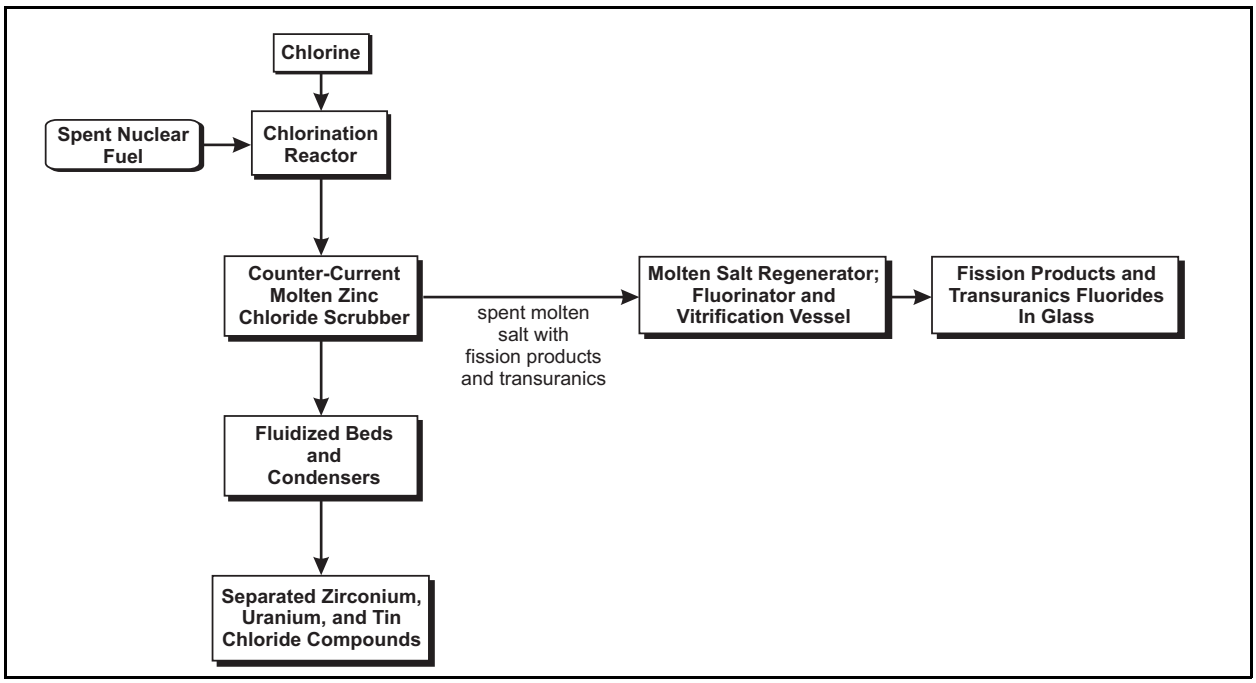

Figure 2–9 Chloride Volatility Process Flow Diagram

2.3.8 Direct Disposal

For the purpose of this EIS, direct disposal of sodium-bonded spent nuclear fuel is disposal without sodium removal. The sodium-bonded spent nuclear fuel (driver and blanket) would be packaged in high-integrity cans as described in Section 2.3.3 without removing the metallic sodium. The high-integrity cans would be placed into a standardized canister designed to provide containment under repository conditions during preclosure operations. At the present time, RCRA mixed waste (which contains both hazardous and radioactive waste) does not meet the requirements of acceptable waste as identified in the current April 1999 *Civilian Radioactive Waste Management System-Waste Acceptance System Requirements Document*, (DOE 1999a). Because of the presence of metallic sodium, the sodium-bonded spent nuclear fuel could be categorized as a RCRA hazardous waste that is potentially both pyrophoric and reactive (DOE 1997). Additionally, the NRC prohibits the disposal of materials that contain or generate explosive, pyrophoric, or chemically reactive substances that could compromise the repository's performance. Therefore, direct disposal would not meet current DOE or NRC repository acceptance criteria.

2.3.9 Sodium Removal and Disposition

As discussed in Section 2.2 and the preceding sections, the disposition of the metallic sodium in the sodium-bonded spent nuclear fuel varies with the treatment method. For those methods that do not require the removal of metallic sodium prior to treatment, or decladding of the fuel (e.g., the electrometallurgical process), the metallic sodium would be converted into a nonreactive salt as part of the process and would be incorporated in the high-level radioactive waste product of the process.

For the methods that require the removal of sodium prior to treatment and/or decladding of the spent fuel (i.e., the PUREX process, the melt and dilute process for blanket spent nuclear fuel [Options 1 and 2], and the packaging in high-integrity cans) the removed metallic sodium would be processed separately, converted into a nonreactive salt, and disposed of as low-level radioactive waste or high-level radioactive waste along with the waste form associated with the treatment process. Decladding and sodium removal could be done using either a mechanical process (the melt, drain, evaporate, and calcinate [MEDEC] process) or a laser declad and alcohol wash process.

In the MEDEC process, the blanket fuel is brought into an argon-atmosphere hot cell where the ends of the cladding for each fuel element are cut off to expose the sodium within the cladding. An argon-atmosphere is required for work involving materials such as sodium which could react with the moisture in air. Then the fuel is placed into a crucible furnace where it is subjected to a temperature of about 200 °C (390 °F), causing melting of the sodium, which is drained into a collection tank. After this bulk sodium is removed, the fuel temperature is raised to about 500 °C (930 °F) and a 10^{-4} torr vacuum is applied to the chamber housing this fuel. This higher temperature vacuum step volatilizes residual sodium and removes it from the fuel. This vacuumed sodium vapor would be condensed in a trap and collected with the previously removed bulk sodium pending further processing.

Sodium recovered during the cleaning process may contain some fission products, most notably cesium-137. The sodium would be stabilized using an oxidation/carbonation process (ANL 1999). Under this process, the cesium would be recovered by vacuum distillation of the sodium, taking advantage of the large difference in the boiling points of the two elements. The boiling point of cesium is 690 °C (1,274 °F), while the boiling point of sodium is 892 °C (1,638 °F). A vapor trap would be placed between the distillation column and pump to collect volatile species emitted from the condenser. The purified sodium would be processed by injection into a chamber, where it would react rapidly with oxygen and water to form aqueous sodium hydroxide. Carbon dioxide gas then would be bubbled through the hydroxide solution converting the sodium hydroxide to sodium carbonate. The aqueous sodium carbonate would be solidified with a binder and

packaged for disposal as low-level radioactive waste. The cesium fraction collected as distillate from the separation process would be added to the high-level radioactive waste form from the process.

To remove the cladding after sodium has been extracted, a special machine would be installed in the argon cell. This machine would mechanically push the fuel pins within the cladding out through the opening created when the cladding ends of the fuel elements previously were cut off. Experience with unirradiated blanket fuel at Argonne National Laboratory has shown that the pins could be mechanically pushed out of the stainless steel cladding after the sodium has been eliminated.

For the melt and dilute process for driver spent nuclear fuel (Option 3), the sodium removed prior to the process would be processed separately, converted into a nonreactive salt, and disposed of as low-level radioactive waste. Any sodium remaining within the fuel would be removed during the melt and dilute process as nonreactive salt, stabilized in a ceramic waste form, and disposed of as high-level radioactive waste.

The laser declad and alcohol wash process has been performed at Rockwell International Hot Laboratory in Canoga Park, California. The activities in this process would be similar to those that were performed at Rockwell International. The process would use a modified laser system for remote operation and a cutting machine to hold and index the fuel elements. The fuel elements would be brought into a hot cell and cut in a predefined cutting sequence. The fumes generated during the cutting process would be filtered and exhausted through an off-gas system. The fuel pins along with the cladding strips would be washed in an alcohol/water mixture to neutralize the metallic sodium and fission product (i.e., cesium) contamination. The fuel pins would be packaged and stored, or sent to SRS for treatment. The alcohol/water solution would be partially evaporated, and the sodium/cesium alcoholates and hydroxides would be neutralized, then solidified in a grouting agent, and disposed of as low- or high-level radioactive waste, depending on the cesium content.

Several aspects of the Rockwell laser process would not meet current environmental standards and would violate the design requirements of an argon hot cell. First, the Rockwell laser process required personnel entry into the hot cell on a biweekly basis for laser maintenance and purging of the cell atmosphere to maintain a low oxygen level (less than 4 percent) and to vent alcohol/water vapors and hydrogen gas from the cell. Neither of these practices would be acceptable for argon cell operation today, in part because of stricter radiation exposure controls and a higher concentration of fission products in the remaining inventory of EBR-II blanket fuel relative to the fuel that was treated by Rockwell. Operation of an argon cell requires maintenance of a low moisture and low oxygen content atmosphere as well as limitations on liquids within the cell for criticality control. The alcohol wash process introduces a liquid which is evaporated into the cell. Second, sodium collected during previous laser decladding operations was able to be disposed of as low-level radioactive waste. The sodium collected from processing the fuel addressed by this EIS would be contaminated with cesium. If sufficient quantities of cesium were present in the sodium, this waste could not be managed as low-level radioactive waste. For the sodium to be managed as low-level radioactive waste, the sodium would have to be processed (as is done with the sodium removed from the fuel in the MEDEC process) using a currently undefined process to remove the cesium from the alcohol mixture. While criticality concerns related to high moisture content levels within a multipurpose argon cell could be eliminated by removing any stored fissile materials, frequent purging of the hot cell atmosphere and personnel entry would still be restricted by current radiation exposure controls and the high concentration of fission products involved. Only the MEDEC process was used to evaluate the various alternatives that require cleaning and/or decladding of the sodium-bonded spent nuclear fuel because of compatibility concerns about laser operation in the cell.

Table 2–3 a summarizes sodium removal and disposition for the treatment methods addressed in this EIS.

Table 2–3 Sodium Removal and Disposition by Treatment and Management Method

Treatment and Management Methods	*Decladding*	*Sodium Treatment*	*Sodium Disposition*
Electrometallurgical process Driver and blanket fuel	No	Stabilization	Converted into nonreactive form, as part of the process, and disposed of with the high-level radioactive ceramic waste product of the process.
High-integrity cans Blanket fuel	No	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
PUREX process Blanket fuel	Yes	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
Melt and dilute process Driver fuel	No	Removal	Part of the sodium is converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste. The remaining sodium is separated during the process, converted to nonreactive ceramic waste form, and disposed of as high-level radioactive waste.
Blanket fuel	Yes [a]/No [b]	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
Direct disposal [c] Driver and blanket fuel	No	No	Disposed of in metallic reactive form in high-integrity cans.

[a] Melt and dilute process at SRS.
[b] Melt and dilute process at ANL-W.
[c] The direct disposal option may not meet current NRC and/or RCRA requirements.

2.4 SPENT NUCLEAR FUEL TREATMENT AND MANAGEMENT FACILITIES

For each alternative, DOE would use existing spent nuclear fuel management facilities that provide remote-handling and heavy-lifting capability, hot cells, and space to receive sodium-bonded spent nuclear fuel shipments. These facilities would prepare, treat, and/or place the sodium-bonded spent nuclear fuel in interim storage awaiting treatment as needed. Besides treating the sodium-bonded spent nuclear fuel, these facilities would provide capabilities to open the shipping containers, sample and analyze the fuel, and vacuum-dry the spent nuclear fuel. These facilities also could be used to repackage the fuel into storage canisters and place the repackaged fuel in dry interim storage to await treatment. The spent nuclear fuel management facilities described in the following sections provide the capability to implement the proposed action for each of the previously described technology alternatives.

2.4.1 ANL-W

The ANL-W site is a center of nuclear technology development and testing (**Figure 2–10**). Five nuclear test reactors have operated on the site, although the only one currently active is a small reactor used for radiography examination of experiments, waste containers, and spent nuclear fuel. Work on highly radioactive materials is conducted in the Fuel Conditioning Facility and the Hot Fuel Examination Facility, both heavily shielded hot cell facilities. Inventories of nuclear materials are maintained on site for conducting research, as well as for storage, pending decisions for further disposition.

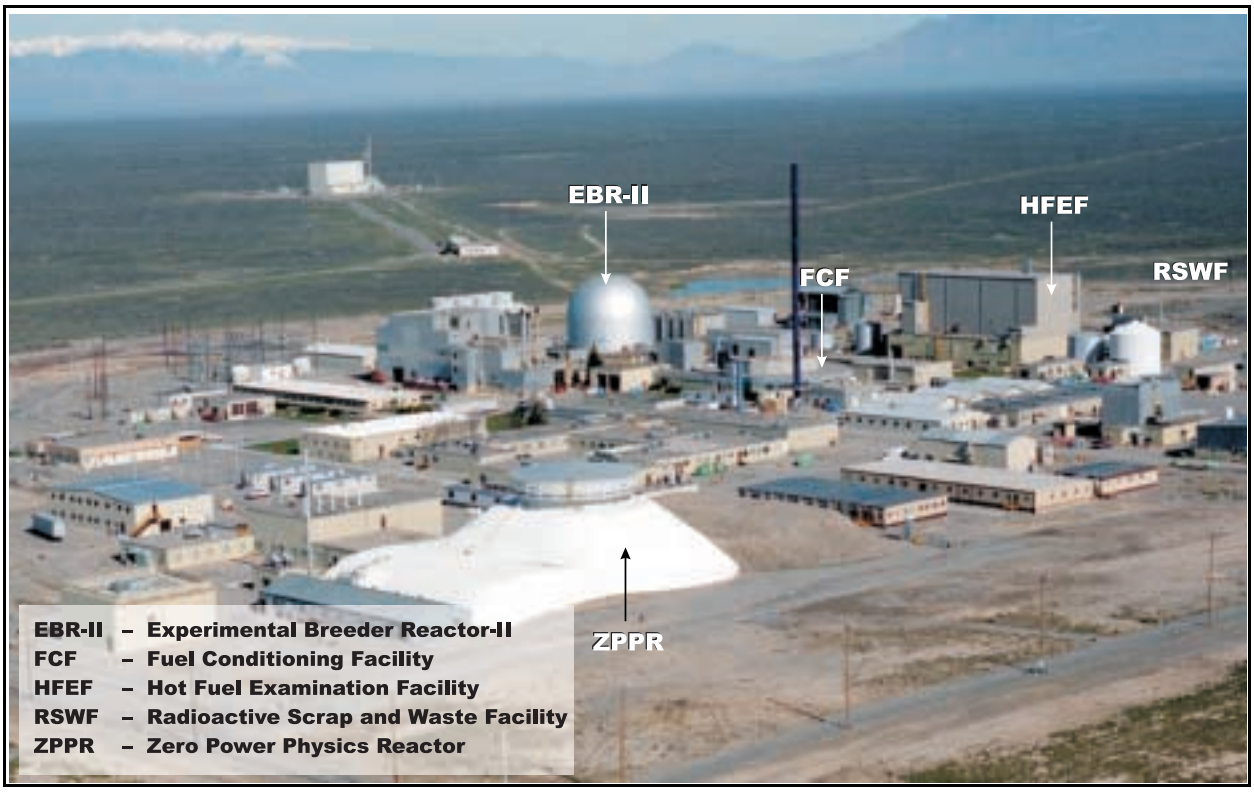

Figure 2–10 ANL-W

2.4.1.1 Fuel Conditioning Facility

The Fuel Conditioning Facility is one of the proposed facilities for the treatment and management of sodium-bonded spent nuclear fuel. The Fuel Conditioning Facility was activated in 1963 and consists of two hot cells, one with an air atmosphere and the other with an inert argon gas atmosphere. Since 1990, the Fuel Conditioning Facility has undergone major reconstruction and refurbishment to meet current safety and environmental requirements. A photographic view of the Fuel Conditioning Facility is shown in **Figure 2–11**. The hot cells enable technicians to work safely with radioactive nuclear materials from behind 1.5-meter-thick (5-foot-thick) shielding walls. A schematic plan of the main floor of the Fuel Conditioning Facility is shown in **Figure 2–12**. The rectangular air cell is used for handling, storage, and assembly/disassembly of components. The argon cell is a much larger hot cell and is "doughnut"-shaped; that is, personnel can work from the outside corridor around the hot cell and can monitor the work in the hot cell from an inner shielded work space in the center of the hot cell.

All equipment in the cells can either be repaired remotely using externally controlled robotic arms (manipulators) and cranes or can be removed and decontaminated for repair elsewhere in the facility from outside corridors around the hot cells.

In addition, the facility contains a mockup area where new equipment can be qualified and tested for remote operation and maintenance prior to installation in the cells. There is also a spray chamber, special glove boxes, and a suited-entry repair area (located in the basement) where contaminated equipment can be decontaminated and repaired.

Figure 2–11 Fuel Conditioning Facility at ANL-W

2.4.1.2 Hot Fuel Examination Facility

The Hot Fuel Examination Facility is one of the proposed facilities for the treatment and management of sodium-bonded spent nuclear fuel. The Hot Fuel Examination Facility is a hot cell complex built in the early 1970s for the preparation and examination of irradiation experiments in support of a wide variety of programs and process demonstrations. A photographic view of the Hot Fuel Examination Facility is shown in **Figure 2–13**. A wide range of remote operations and examinations may be performed in this facility with its shielded cells, support areas, and equipment.

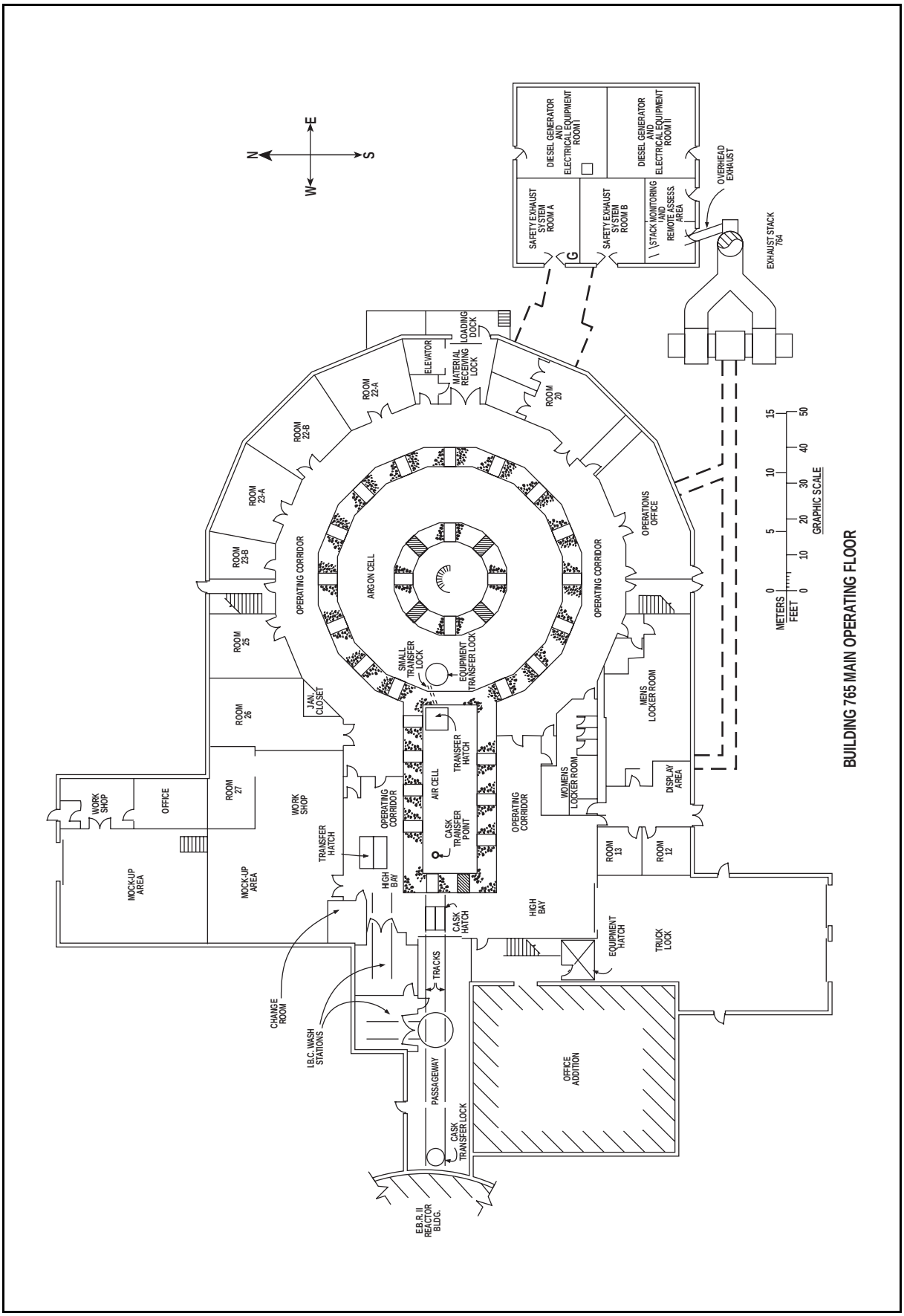

Figure 2–12 Main Floor Layout of the Fuel Conditioning Facility at ANL-W

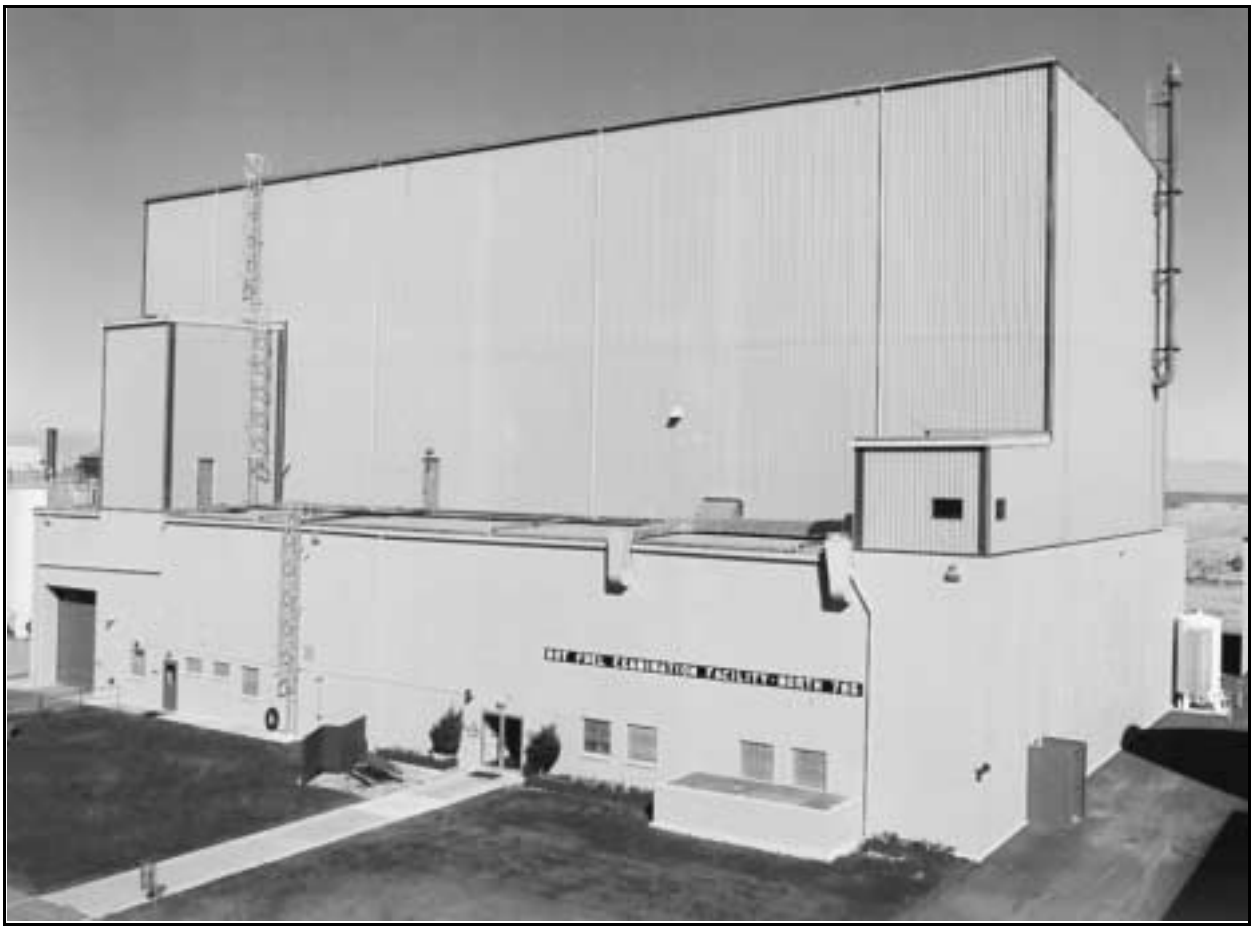

Figure 2–13 Hot Fuel Examination Facility at ANL-W

The Hot Fuel Examination Facility is designed to be adapted to a wide variety of programs and consists primarily of two adjacent shielded cells, the main cell and the decontamination cell, in a three-story building. The decontamination cell contains an air atmosphere. A schematic plan of the main floor of the Hot Fuel Examination Facility is shown in **Figure 2–14.** The main cell contains an argon atmosphere for work involving materials such as sodium, plutonium, and other materials which could react with air. Both cells are surrounded by 1.2-meter-thick (4-foot-thick), high-density concrete to protect workers from the high radiation levels present in the hot cells. There are 21 work stations in the Hot Fuel Examination Facility, all equipped with shielded windows and remote manipulators. All in-cell equipment is carefully designed to permit remote operation and maintenance. A truck lock is located at the west end of the cell complex. The truck lock is large enough to accommodate various trucks and fork lifts which are used to transport shielded casks used in the day-to-day operation of the facility. The facility has recently been modified to accept truck-sized spent nuclear fuel shipping casks.

A high bay area covering the entire cell complex and serviced by a 40-ton bridge crane provides access from the truck lock to the top of the cells for bottom-opening casks. This area contains repair rooms, change rooms, and an access room and provides space for clean equipment repair and mockup.

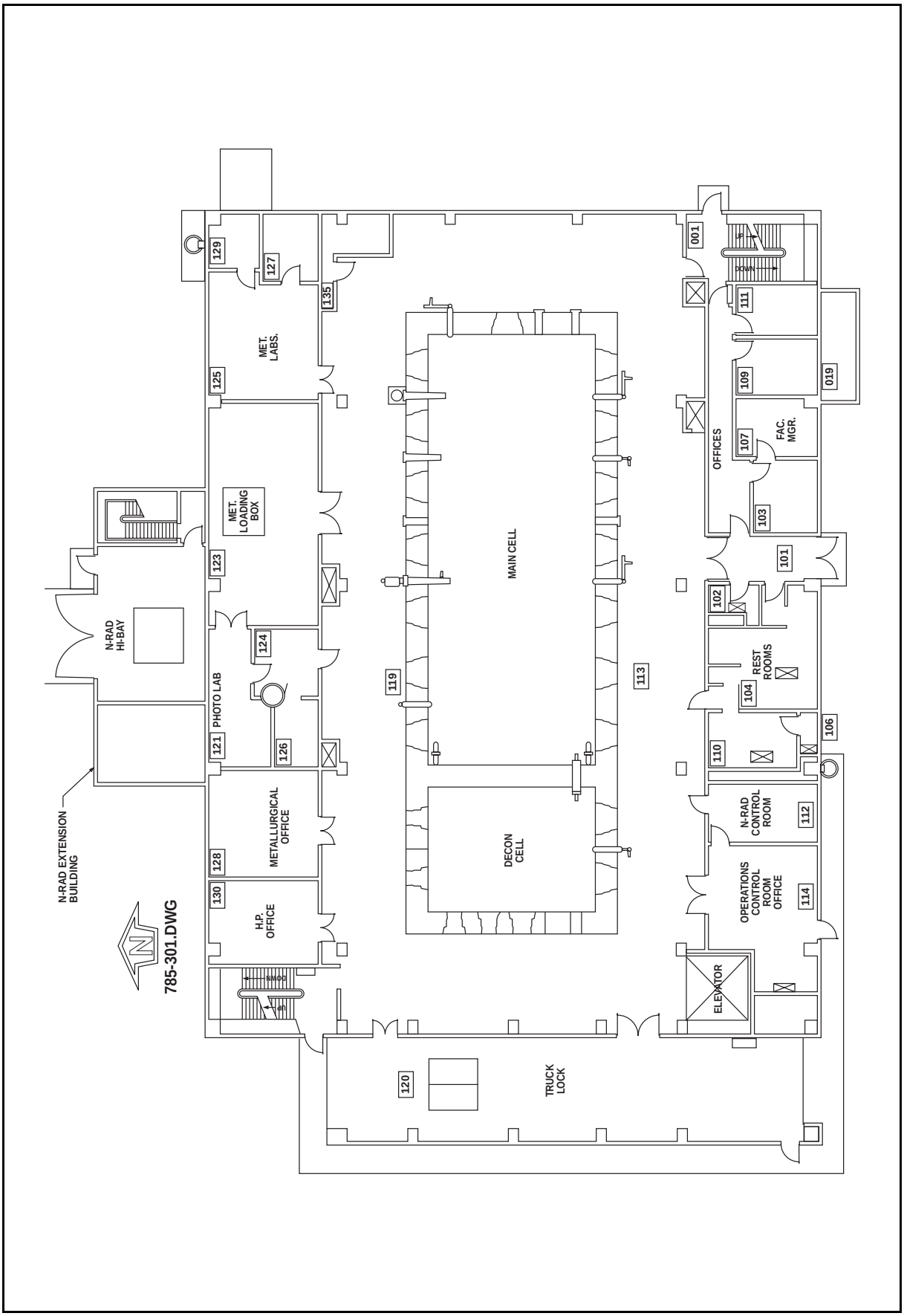

Figure 2–14 Main Floor Layout of the Hot Fuel Examination Facility at ANL-W

2.4.1.3 Zero Power Physics Reactor Materials Storage Building

The Zero Power Physics Reactor is currently shut down, but the facility is used for a number of projects, including a gas generation experiment. The Zero Power Physics Reactor Materials Storage Building, shown in **Figure 2–15**, is one of the primary storage facilities at ANL-W for uranium metal. Inventories of nuclear materials stored in this facility are maintained for conducting research as well as for storage, pending decisions for further disposition.

Figure 2–15 Zero Power Physics Reactor Materials Storage Building at ANL-W

2.4.1.4 Radioactive Scrap and Waste Facility

The Radioactive Scrap and Waste Facility, occupying about 1.6 hectares (4 acres), provides safe interim dry storage for spent nuclear fuel and waste generated from experiments. It is one of the facilities where sodium-bonded spent nuclear fuel currently is stored and the facility where high-level radioactive waste from the treatment of the fuel could be stored pending ultimate disposal. It is located underground and 0.8 kilometers (0.5 miles) northeast of the main ANL-W facilities within the ANL-W boundary. The Radioactive Scrap and Waste Facility looks somewhat like a large parking lot on the surface, as shown in **Figure 2–16**. The facility has a permit issued by the State of Idaho for interim storage of mixed waste regulated under RCRA. The Radioactive Scrap and Waste Facility provides protection against corrosion for the more than 1,000 underground steel liners available for waste storage of materials handled at ANL-W.

Figure 2–16 Radioactive Scrap and Waste Facility at ANL-W

2.4.2 INTEC at INEEL

INTEC is located 20 kilometers (12.4 miles) west-southwest of ANL-W. It is one of the sites where sodium-bonded spent nuclear fuel currently is stored. A photographic view of INTEC is shown in **Figure 2–17**. INTEC was constructed in the 1950s to reprocess spent nuclear fuel from government reactors. In 1992, DOE announced that it no longer would reprocess spent nuclear fuel. Current work at INTEC includes receiving and storing spent nuclear fuel, converting liquid sodium-bearing waste to granular solid, environmental restoration and decontamination and dismantling activities, and technology development. About 880 people currently work at INTEC. The facility would be used to continue storing sodium-bonded spent nuclear fuel and for packaging the treated sodium-bonded spent nuclear fuel in standardized canisters in preparation for transport and disposal in a geologic repository. However, because it has no hot cell with an inert gas atmosphere, it cannot be used for any sodium removal activities under the proposed action.

Figure 2–17 INTEC at INEEL

The primary facilities at INTEC include:

- The Fluorinel Dissolution Process and Fuel Storage Facility. This facility is divided into two parts, a spent nuclear fuel storage area and the Fluorinel Dissolution Facility. The storage area consists of six storage pools for storing nuclear fuel. Radioactive spent nuclear fuel is stored under about 11 million liters (3 million gallons) of water, which provide protective shielding and cooling. Eventually, all spent nuclear fuel will be removed from underwater storage pools and placed in a dry storage system and prepared for shipment to a repository.

  The spent nuclear fuel, from government-owned reactors, was formerly reprocessed at INTEC to recover reusable uranium. The Fluorinel Dissolution Facility includes an air atmosphere "hot cell" with 1.8-meter-thick (6-foot-thick) concrete walls where spent nuclear fuel was dissolved in an acid solution. With the end of reprocessing, uranium and hazardous materials were flushed from the Fluorinel Dissolution Facility. New missions for this facility are under consideration.

- The 603 Fuel Storage Building. This building houses both underwater pools and dry storage facilities for spent nuclear fuel. The pools were constructed in the 1950s and served as the primary spent nuclear fuel storage facility until the Fluorinel Dissolution Process and Fuel Storage Facility opened in 1984. Fuel in underwater storage at Building 603 is being transferred to the newer storage pools at the Fluorinel Dissolution Process and Fuel Storage Facility. Also located in the building is the Irradiated Fuel Storage Facility, which stores dry fuel that cannot be stored underwater. The Irradiated Fuel Storage Facility has 636 storage positions, with 297 in use. The majority of the spent nuclear fuel stored in this facility came from the Fort St. Vrain commercial reactor in Colorado.

- The New Waste Calcining Facility. This facility converted liquid high-level radioactive waste into a granular solid similar in consistency to dry laundry detergent. The liquid waste was drawn from underground storage tanks and sprayed into a vessel superheated by a mixture of kerosene and oxygen. Most of the liquid evaporated, while radioactive fission products adhered to the granular bed material in the vessel. The off-gases were treated and monitored before they were released to the environment, and the residual solids were transferred to large stainless steel structures encased in thick concrete vaults. This conversion process achieved an 8-to-1 volume reduction from liquid to solid. The same process currently is used to convert sodium-bearing waste to granular solid with a smaller volume reduction from liquid to solid.

- The Remote Analytical Laboratory. This laboratory is designed for the safe examination of radioactive samples to support the New Waste Calcining Facility mission and other INTEC operations. The facility includes a 356-square-meter (3,500-square-foot) air atmosphere hot cell with 90-centimeter (3-foot) concrete shield walls that allow remote examination of radioactive chemical samples.

- The INTEC 601/602 Processing Corridors. The processing corridors were used to extract highly enriched uranium from dissolved spent nuclear fuel during reprocessing and to solidify the recovered highly enriched uranium for shipment off site. Built in the 1950s, these facilities were to be replaced by the Fuel Processing Restoration Project. Because DOE decided to discontinue reprocessing, these facilities have been flushed to remove uranium and hazardous materials. The Fuel Processing Restoration Project, about 40 percent complete when construction stopped in 1992, was discontinued in a manner that preserves the facility for possible use in future research and development missions at INTEC.

- The High-Level Radioactive Waste Tank Farm. The waste tank farm includes 11 underground stainless steel storage tanks used to store the radioactive liquid waste generated during the reprocessing of spent

nuclear fuel and plant decontamination work. DOE is evaluating options for treating the remaining Tank Farm liquid waste in the *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement*, which was issued for public comment in December 1999 (DOE 1999b). The underground tanks are encased in concrete vaults which have sumps and leak detectors. One tank is always kept empty for use as a transfer backup should a problem develop with one of the other 10 tanks. The tanks are corrosion-resistant, and no leakage has been detected. Some leaks from transfer lines outside the tanks have occurred, however, and the contaminated soil is scheduled for environmental cleanup.

2.4.3 SRS

SRS was constructed during the early 1950s to produce the basic materials used in the fabrication of nuclear weapons, primarily tritium and plutonium-239. Five reactors were built on the site. The reactors produced nuclear materials by irradiating target materials with neutrons. Also built were support facilities including two chemical separation plants, a heavy water extraction plant, a nuclear fuel and target fabrication facility, and waste management facilities.

Irradiated materials were moved from the reactors to the two chemical separation facilities—the next step in the production process. In these facilities, known as "canyons," the irradiated fuel and target assemblies were processed chemically to separate useful products from waste. After refinement, some nuclear materials were shipped to other DOE sites for final use.

SRS has adjusted through the years to meet changing defense requirements. All five of the original SRS production reactors are permanently shut down. While production of new tritium will not be necessary for several years, recycling and reloading of tritium to maintain nuclear weapons' reliability is a continuing site mission.

DOE currently uses the F-Canyon chemical separation facility and the FB-Line to stabilize spent nuclear fuel, as described in the *Final Environmental Impact Statement, Interim Management of Nuclear Materials* (DOE 1995b). The *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000) will be used to help determine the most appropriate final disposition option for spent nuclear fuel currently assigned to SRS.

Weapons material production at SRS has produced unusable by-products such as high-level radioactive waste. The high-level radioactive waste, approximately 35 million gallons, is stored in waste tanks on site. The Defense Waste Processing Facility will bond the radioactive elements in borosilicate glass, a stable form for disposal.

2.4.3.1 F-Canyon

The F-Canyon at SRS could chemically separate uranium from fission products in blanket spent nuclear fuel using the PUREX process. A photographic view of the F-Canyon complex is shown in **Figure 2–18**. The canyon facilities use radiochemical processes for the separation and recovery of plutonium and uranium isotopes. Historically, F-Canyon recovered plutonium-239 and uranium-238 from irradiated natural or depleted uranium.

The Canyon building is a reinforced concrete structure, 254 meters (835 feet) long by 37 meters (122 feet) wide by 20 meters (66 feet) high. It houses large equipment (tanks, process vessels, evaporators, etc.) used in the chemical separation processes.

Figure 2–18 The F-Canyon Complex at SRS

The F-Canyon facility actually contains two canyons, the hot canyon and the warm canyon, as shown in **Figure 2–19**. The two canyons are parallel and separated by a center section, which has four floors. The center section contains office space, the control room for facility operations, chemical feed systems, and support equipment such as ventilation fans. Processing operations involving high radiation levels (dissolution, fission product separation, and high-level radioactive waste evaporation) occur in the hot canyon, which has thick concrete walls to shield people outside and in the center section from radiation. The final steps of the chemical separation process, which generally involve lower radiation levels, occur in the warm canyon.

2.4.3.1.1 FB-Line

The FB-Line, located on top of the F-Canyon, historically converted plutonium nitrate solution produced in the F-Canyon to plutonium-239 metal buttons. Solutions from the F-Canyon are concentrated and purified in the FB-Line. The plutonium then is precipitated, filtered, dried, and finally reduced to a metallic form called a button. The button is about the size of a hockey puck. Processing equipment is enclosed in glove boxes so that employees and operating areas are not exposed to the radioactive material. Some of these operations are automated. The FB-Line also recycles plutonium scrap generated from facility operations and offsite sources. In September 1997, the FB-Line began a new plutonium packaging process. This process places stabilized plutonium in rugged, welded stainless steel cans. DOE also has determined that the FB-Line should be used to stabilize plutonium.

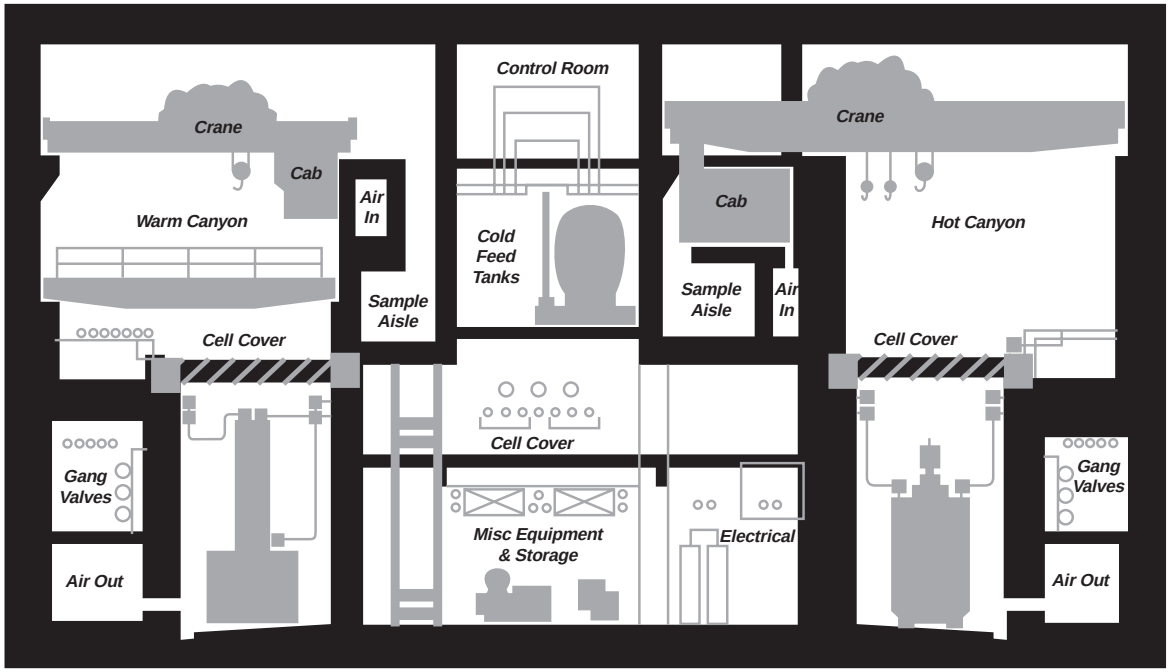

Figure 2–19 F-Canyon Building Sections (Hot Canyon and Warm Canyon) at SRS

2.4.3.2 Building 105-L

Building 105-L is the SRS facility where installation of a melt and dilute process for treating spent nuclear fuel is proposed. Building 105-L is part of the currently shut-down L-Reactor complex at SRS. The L-Reactor was built in the early 1950s to produce nuclear materials for national defense. In 1988, DOE shut the reactor down for safety upgrades. At the completion of the upgrades, the reactor was not restarted. A photographic view of Building 105-L is shown in **Figure 2–20**. In 1993, DOE ended the reactor's materials production mission. The current mission of this facility is to store reactor components and other radioactive materials in the disassembly basin; receive and store foreign and domestic research reactor fuel in the disassembly basin; decontaminate shipping casks in the Building 105-L stack area; store contaminated moderators in tanks or drums; and compact low-level radioactive waste in a compactor. DOE maintains the structures, systems, and components necessary to perform these missions, but has de-energized, drained, or otherwise deactivated many others.

Building 105-L has space potentially suitable for installation of a melt and dilute process (DOE 2000). The space includes the process room and crane maintenance area. The process room, a shielded area situated above the reactor tank, formerly provided access to the reactor by means of a charge and discharge machine for handling reactor fuel assemblies. An overhead crane services the area. The crane maintenance area, connected to the process room by a shielded crane wash area, allows hands-on maintenance of the fuel assembly transfer systems. The Building 105-L stack area would be used to unload shipping casks from their containers and to decontaminate empty shipping casks.

In the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement*, (DOE 2000), DOE identified melt and dilute as one of the preferred methods of treating spent nuclear fuel at SRS. To implement the melt and dilute technology, DOE would construct a melt and dilute facility in the existing Building 105-L at SRS and build a dry storage facility in L-Area, near Building 105-L.

Figure 2–20 L-Reactor Complex at SRS

DOE expects the melt and dilute option would be relatively simple to implement in Building 105-L. The major technical issue in implementing this technology would be the design of an off-gas system to capture volatilized fission products. Preliminary engineering studies indicate that the system could be designed using proven approaches for managing off-gases. The impacts from the construction of a melt and dilute facility at SRS's Building 105-L are addressed in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000).

2.4.3.3 Defense Waste Processing Facility

The Defense Waste Processing Facility, located in the S-Area, converts high-level radioactive liquid waste currently stored at SRS into a solid glass form suitable for long-term storage and disposal. A photographic view of the Defense Waste Processing Facility is shown in **Figure 2–21**. This process, called "vitrification," immobilizes high-level radioactive liquids into a more stable form suitable for disposal in a geologic repository. About 125 million liters (34 million gallons) of high-level liquid radioactive waste currently is stored in 49 underground carbon steel tanks at SRS. This waste has about 480 million curies of radioactivity, and requires permanent isolation from the environment.

Figure 2–21 Defense Waste Processing Facility at SRS

Construction of the Defense Waste Processing Facility began in 1983. Changing environmental requirements; major safety upgrades and process modifications; and a "waste qualification" test to demonstrate that the glass form meets all environmental and operational requirements for long-term storage were required before system testing began in 1990. The Defense Waste Processing Facility successfully completed its waste qualification testing in late 1995 and began operating in March 1996.

The Defense Waste Processing Facility treats the highly radioactive material removed from the original waste. In this process, a sand-like borosilicate glass is mixed with the waste and sent to the plant's steel and ceramic melter. In the melter, electricity is used to heat the waste/borosilicate glass mixture until molten. This molten glass-waste mixture is poured in a pencil-thin stream into stainless steel canisters to cool and harden. Each canister is approximately 3 meters (10 feet) tall and 0.6 meters (2 feet) in diameter; it takes approximately 24 hours to fill one canister. A filled Defense Waste Processing Facility canister weighs about 2,270 kilograms (5,000 pounds). The exterior of each canister is blasted with borosilicate glass to remove contamination, then welded shut after a plug has been rammed into place.

A specially designed "Shielded Canister Transporter" moves each sealed canister, one at a time, from the Defense Waste Processing Facility to the temporary storage building adjacent to the facility. This transporter is a two-wheel drive vehicle powered by redundant diesel engines. It has a center module with a shielding cask, floor plug cavity, and associated cask lifting equipment.

At the storage building, canisters are lowered by the transporter into an underground reinforced concrete vault containing 2,286 individual canister supports. This seismically qualified storage vault can hold, at current Defense Waste Processing Facility production flow, canisters from about 8 to 10 years of processing. More storage

buildings can be built according to the need for storage space. The canisters are stored at SRS until a Federal repository can be established.

2.5 ALTERNATIVES EVALUATED

As discussed in Section 2.1, the proposed action is to treat and manage sodium-bonded spent nuclear fuel. The alternatives under the proposed action are illustrated in **Figure 2–22** and are addressed below. Although each alternative addresses both driver and blanket spent nuclear fuel, DOE will consider the driver and blanket spent nuclear fuel, as well as the EBR-II and Fermi-1 blanket spent nuclear fuel, separately in identifying a preferred alternative and any subsequent Record of Decision. In other words, DOE is considering all combinations of technologies, options, and fuel types, not only the specific combinations that are explicitly discussed in the EIS. For example, "no action" may be chosen for the driver spent nuclear fuel and "melt and dilute at SRS" for the blanket spent nuclear fuel.

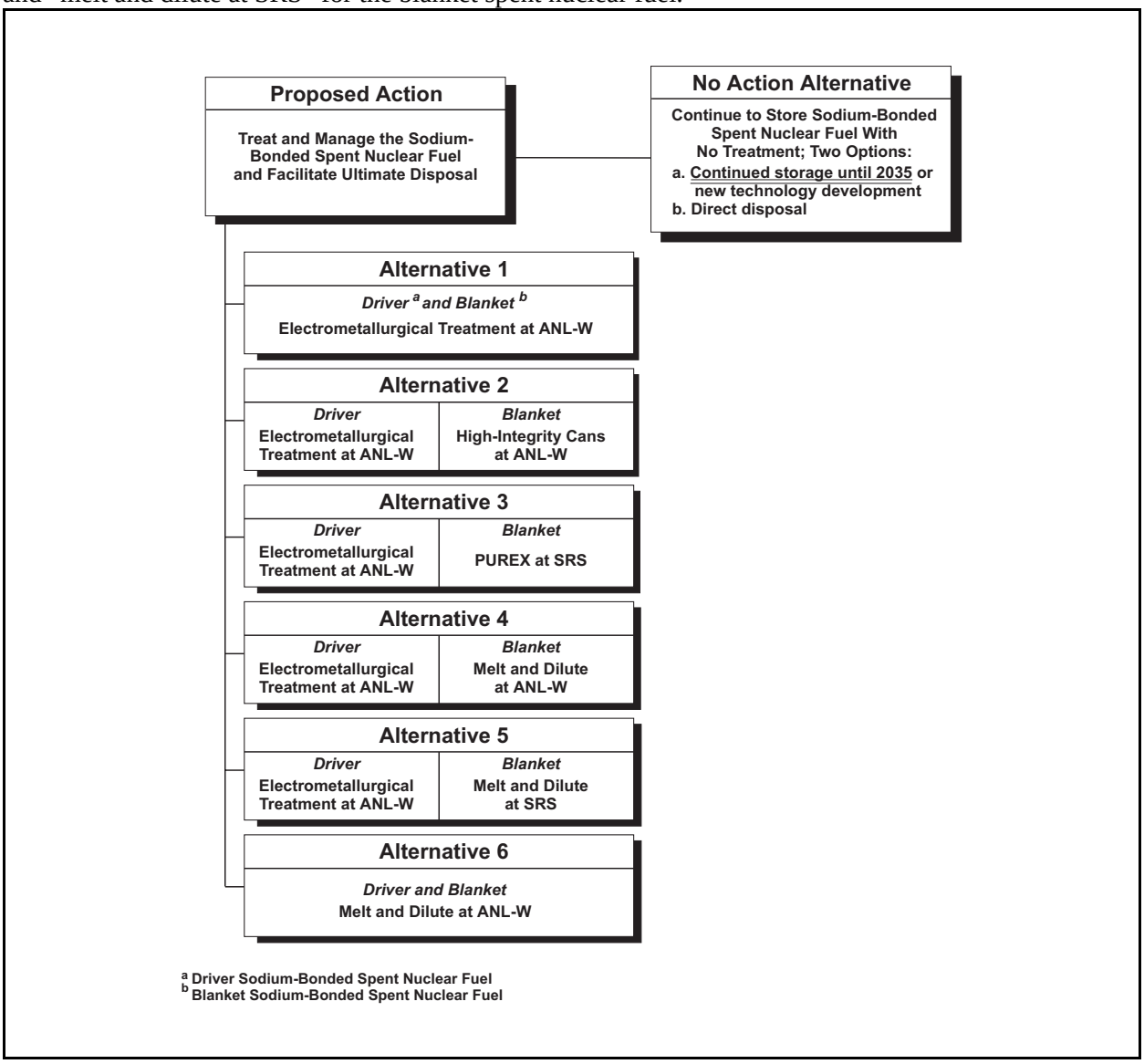

Figure 2–22 Proposed Action and Alternatives

2.5.1 No Action Alternative

Under the No Action Alternative, all or part of the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed), except for stabilization activities that may be necessary to prevent potential degradation of some of the spent nuclear fuel. Under the No Action Alternative, two options were analyzed: (1) the sodium-bonded spent nuclear fuel would continue to be stored until 2035 at its current location subject only to activities dictated by the amended Record of Decision (61 FR 9441) for the Programmatic Spent Nuclear Fuel EIS (DOE 1995a) and other existing site-specific NEPA documentation or until a technology currently dismissed as an unreasonable alternative because it is less mature (e.g., GMODS or plasma arc) is developed; and (2) the sodium-bonded spent nuclear fuel would be disposed of directly in a geologic repository without treatment. The fuel would be packaged in high-integrity cans without sodium removal. Under the latter option, the sodium-bonded spent nuclear fuel at INTEC would be transported to ANL-W for packaging. Both options would require the installation of some new waste handling equipment. As discussed in Section 2.3.8, the direct disposal option would not meet current DOE or NRC repository acceptance criteria requirements.

A fundamental assumption made under the No Action Alternative is that the sodium-bonded spent nuclear fuel eventually will be disposed of in a manner similar to the rest of the spent nuclear fuel owned by DOE and within the time period considered over which institutional controls could reliably be assumed to be in effect. In the event that the sodium-bonded spent nuclear fuel has not been treated before 2035, the temporarily stored fuel will be removed from the State of Idaho by the year 2035. The environmental impact of the removal of untreated sodium-bonded spent nuclear fuel would be evaluated in a separate NEPA document. The continued storage of untreated sodium-bonded spent nuclear fuel in the State of Idaho or elsewhere, beyond time periods for which institutional controls could reliably be assumed to be in effect, could lead to significant impacts to the environment and the health and safety of the public from radioactive releases caused by the gradual degradation of the fuel and its containment.

In selecting the No Action Alternative, DOE could actively pursue research and development of another treatment technology including, for example, the GMODS and plasma arc methods. These methods offer the potential for treating both driver and blanket spent nuclear fuel. They do not involve separation of uranium or plutonium, and the treatment product is expected to be suitable for disposal in a geologic repository. Reasons for not including these methods among the reasonable alternatives under the proposed action are provided in Section 2.6.

2.5.2 Alternative 1: Electrometallurgically Treat Blanket and Driver Fuel at ANL-W

Under this alternative, all sodium-bonded spent nuclear fuel (both driver and blanket, approximately 60 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process. **Figure 2–23** illustrates the steps of the process under Alternative 1.

The sodium-bonded spent nuclear fuel (driver and blanket) from ANL-W (the Radioactive Scrap and Waste Facility and the Hot Fuel Examination Facility) would be transported directly to the Fuel Conditioning Facility for electrometallurgical treatment. Spent nuclear fuel currently stored at INTEC would be transported to the Hot Fuel Examination Facility. This is necessary because only the Hot Fuel Examination Facility at ANL-W is capable of accepting spent nuclear fuel transportation casks. At the Hot Fuel Examination Facility, the fuel would be separated from the assembly hardware and packaged and transferred to the Fuel Conditioning Facility for electrometallurgical treatment. The separated hardware would be packaged and managed as low-level radioactive waste.

After treatment, the low-enriched uranium by-product from the cathode processing would be metal-casted at the Fuel Conditioning Facility and transferred to the Zero Power Physics Reactor Materials Storage Building for storage. The cladding hulls remaining at the anode would be packaged and transferred to the

Hot Fuel Examination Facility for metal casting into high-level radioactive waste and transferred to the Radioactive Scrap and Waste Facility for storage. The electrorefiner salt containing the fission products, sodium, and transuranic elements would be transferred in metal cans back to the Hot Fuel Examination Facility where the ceramic waste would be produced. The ceramic waste cylinders would be packaged and transferred to the Radioactive Scrap and Waste Facility for storage. Implementing this alternative at the Fuel Conditioning Facility and the Hot Fuel Examination Facility would require the installation of some new waste handling equipment at the facilities. Electrometallurgical treatment of the sodium-bonded spent nuclear fuel at ANL-W could start as early as the year 2000, and would require approximately 13 years to process all fuel. Driver spent nuclear fuel alone would require approximately 7 years.

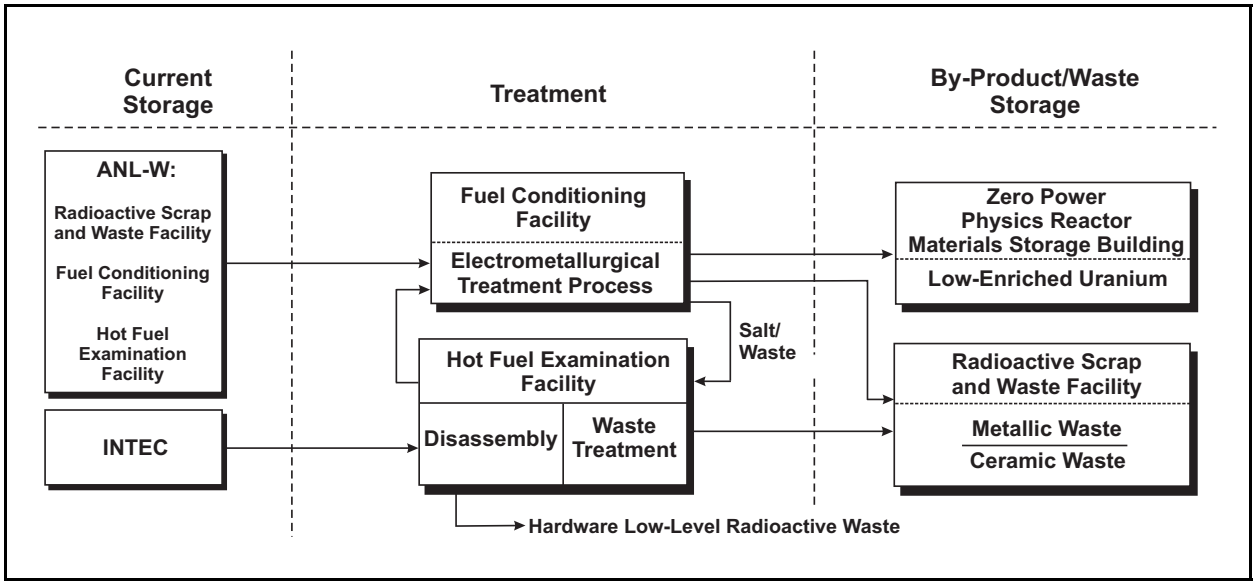

Figure 2–23 Schematic for Driver and Blanket Spent Nuclear Fuel Treatment in Alternative 1

2.5.3 Alternative 2: Clean and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the blanket spent nuclear fuel elements (approximately 57 metric tons of heavy metal) would be packaged in high-integrity stainless steel cans at ANL-W after the sodium has been removed without decladding, as discussed in Section 2.3.9. Removal of the sodium from the sodium-bonded blanket spent nuclear fuel would take place at the Hot Fuel Examination Facility at ANL-W. The packaging in high-integrity cans would take place in the same facility. The high-integrity cans would be transferred to the Radioactive Scrap and Waste Facility for storage. **Figure 2–24** illustrates the steps of the process for the blanket sodium-bonded spent nuclear fuel.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated using the electrometallurgical treatment process as described in Section 2.5.2 for Alternative 1.

Implementing this alternative at the Hot Fuel Examination Facility would require the installation of equipment for sodium removal activities. In addition, some new waste handling equipment would be needed for the electrometallurgical treatment of the driver sodium-bonded spent nuclear fuel.

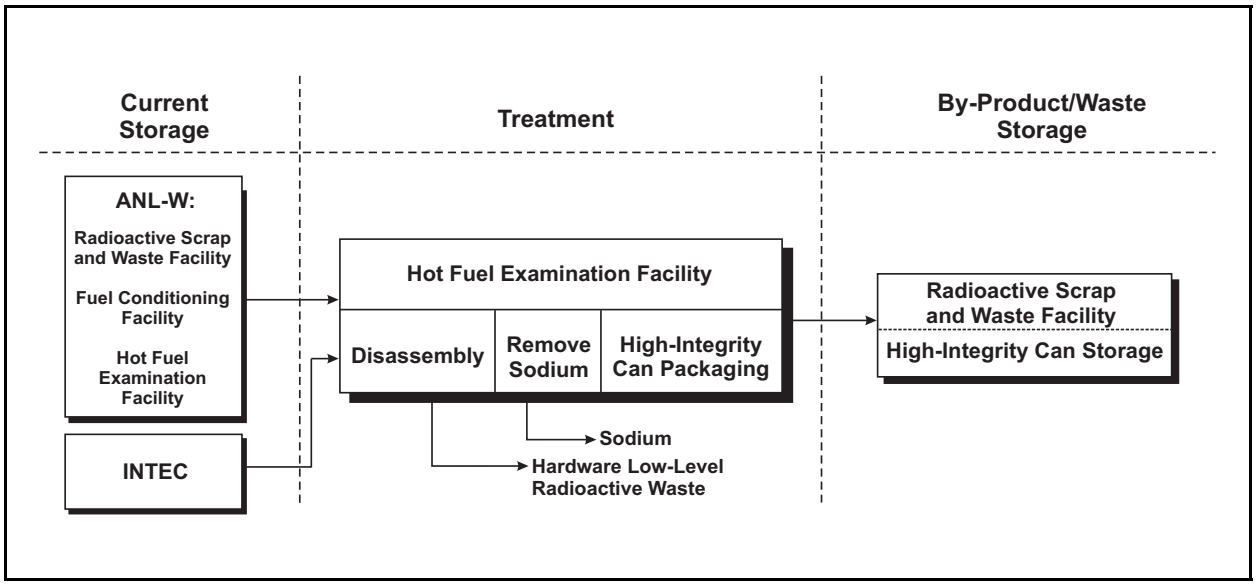

Figure 2–24 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 2

Packaging the blanket spent nuclear fuel in high-integrity cans could start by approximately 2003. It would take approximately six years to complete. Electrometallurgical treatment of the driver spent nuclear fuel would start in 2000 and would be completed in approximately seven years.

2.5.4 Alternative 3: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; PUREX Process Blanket Fuel at SRS

Under this alternative, the blanket spent nuclear fuel pins (approximately 57 metric tons of heavy metal) would be packaged in aluminum cans and shipped to SRS for treatment using the PUREX process at the SRS F-Canyon facility. The blanket spent nuclear fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W, as discussed in Section 2.3.9.

The decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal would take place at the Hot Fuel Examination Facility at ANL-W. Equipment for decladding and sodium removal would need to be installed for this purpose. After decladding and sodium removal, the blanket spent nuclear fuel pins would be packaged and stored temporarily at the Hot Fuel Examination Facility to await shipment to SRS.

At SRS, the cans containing blanket spent nuclear fuel pins would be unpacked at the F-Canyon facility before treatment using the PUREX process. No modifications to that facility would be needed. Waste from the process containing the fission products and transuranic isotopes other than plutonium would be transferred to the Defense Waste Processing Facility where it would be converted to borosilicate glass logs and stored pending ultimate disposal. Approximately 260 kilograms (572 pounds) of separated plutonium would be immobilized using the can-in-canister technology at SRS for eventual geologic repository disposal in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999c). Depleted uranium would be transferred to a storage yard for depleted uranium at the site. **Figure 2–25** illustrates the process steps for the blanket spent nuclear fuel at ANL-W and SRS.

Considering the commitment of F-Canyon to other DOE missions, PUREX processing of the blanket spent nuclear fuel would start no earlier than 2005, and would last less than one year. Decladding and sodium

removal activities at ANL-W would not start earlier than 2003. Therefore, these activities would determine the length of the process.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment processes described in Section 2.5.2 for Alternative 1. As in the case of Alternative 2, electrometallurgical treatment of the driver spent nuclear fuel could start in 2000 and could be completed in approximately seven years. The process steps for the treatment of the driver spent nuclear fuel are shown in Figure 2–23.

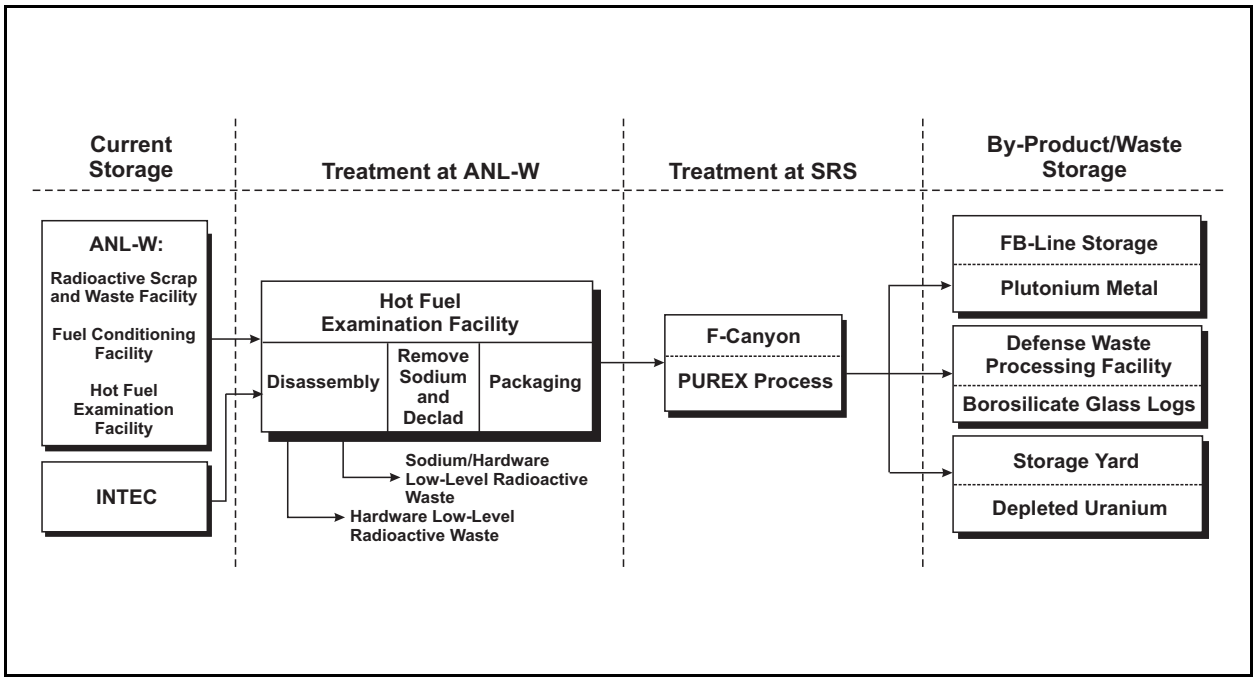

Figure 2–25 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 3

2.5.5 Alternative 4: Melt and Dilute Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the blanket spent nuclear fuel elements (approximately 57 metric tons of heavy metal) would be treated at ANL-W using the melt and dilute Option 2 process described in Section 2.3.4. Prior to treatment, the metallic sodium would be removed without decladding at ANL-W, as discussed in Section 2.3.9. Removal of the sodium from the sodium-bonded blanket spent nuclear fuel could take place at the Hot Fuel Examination Facility at ANL-W. Equipment for sodium removal would need to be installed. Equipment necessary for the melt and dilute process would need to be installed at the Hot Fuel Examination Facility, including the addition of the melter and an off-gas system.

Metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be transferred to the Radioactive Scrap and Waste Facility for storage pending ultimate disposal. **Figure 2–26** illustrates the process steps for the sodium-bonded blanket spent nuclear fuel.

Treatment of blanket spent nuclear fuel at ANL-W using the melt and dilute process could start as early as 2005 and could be completed in eight years.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process described in Section 2.5.2 for Alternative 1. Treatment of the driver spent nuclear fuel could start as early as 2000 and could be completed in approximately seven years. The process steps for the treatment of the driver sodium-bonded spent nuclear fuel are shown in Figure 2–23.

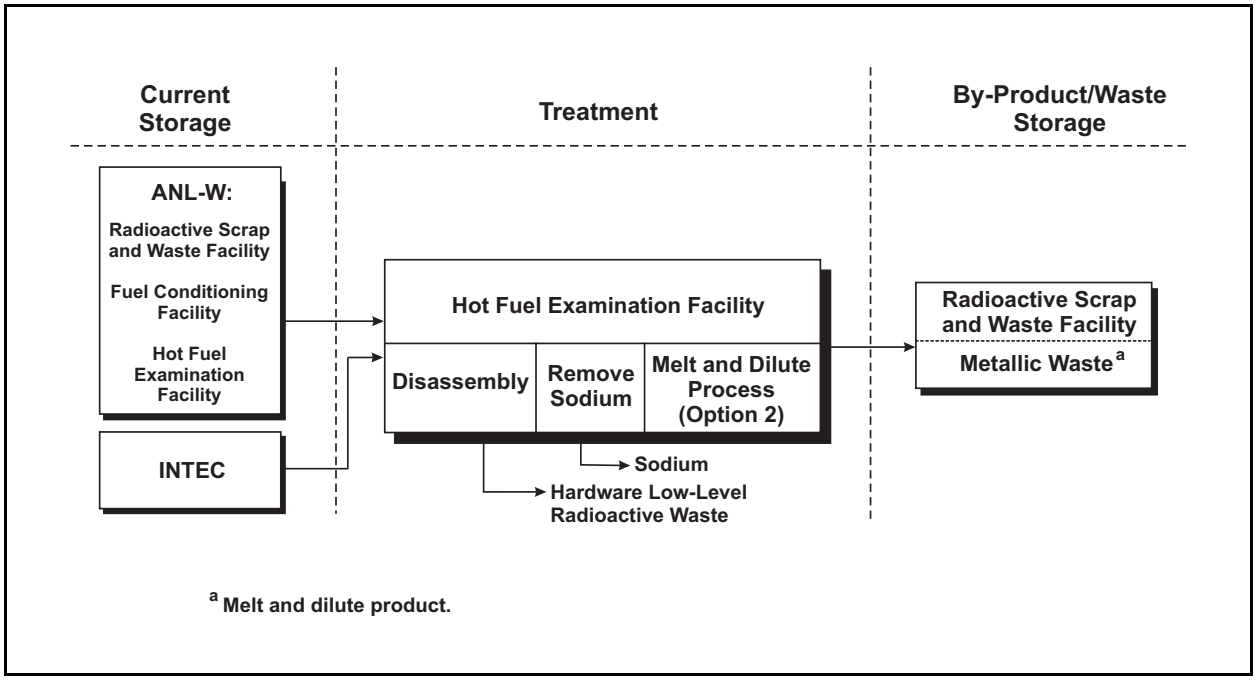

Figure 2–26 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 4

2.5.6 Alternative 5: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; Melt and Dilute Blanket Fuel at SRS

Under this alternative, the blanket spent nuclear fuel pins (approximately 57 metric tons of heavy metal) would be packaged and shipped to SRS for treatment. The blanket spent nuclear fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W. The declad and cleaned blanket spent nuclear fuel pins would be received at Building 105-L at SRS and treated using the melt and dilute Option 1 process, as described in Section 2.3.4.

Decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal would take place at the Hot Fuel Examination Facility at ANL-W, as discussed in Section 2.3.9. Spent nuclear fuel currently stored at ANL-W facilities could be transported directly to the Hot Fuel Examination Facility. After decladding and sodium removal, the blanket spent nuclear fuel pins would be packaged and stored temporarily at the Hot Fuel Examination Facility pending shipment to SRS.

At SRS, the cans containing the blanket spent nuclear fuel pins would be unpacked at Building 105-L and the blanket spent nuclear fuel pins would be treated using the melt and dilute process. For the purpose of evaluating this alternative, it is assumed that the melt and dilute facility is operational at SRS, as proposed in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000).

Metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be stored at the L-Area storage pending ultimate disposal.

Figure 2–27 illustrates the process steps for the blanket spent nuclear fuel at ANL-W and SRS.

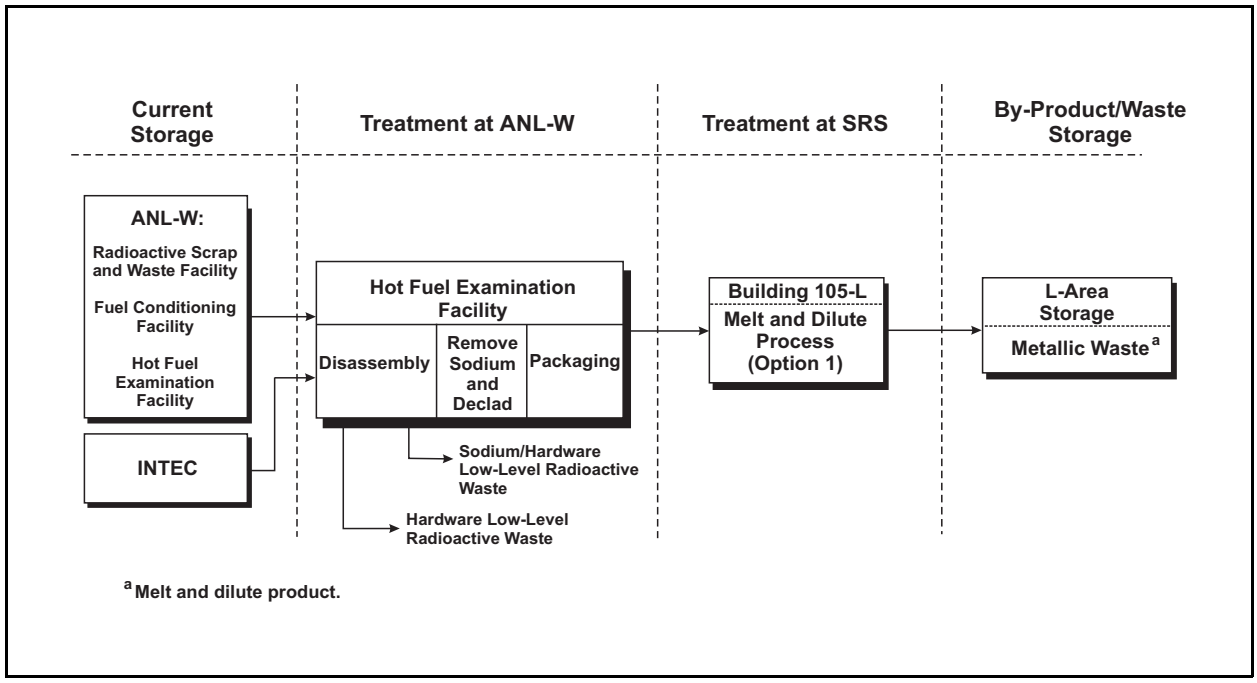

Figure 2–27 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 5

Treatment of the blanket spent nuclear fuel at SRS would start around 2035. The facility would be operational in 2005 and is committed to other DOE missions until 2035. If additional capacity becomes available, treatment could start as soon as 2020. The treatment process would last approximately three years. Until 2035, there would be ample time for blanket spent nuclear fuel decladding and sodium removal activities at ANL-W.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process described in Section 2.5.2 for Alternative 1. Treatment of the driver spent nuclear fuel at ANL-W could start in 2000 and could be completed in approximately seven years. The process steps for the treatment of the driver sodium-bonded spent nuclear fuel are shown in Figure 2–23.

2.5.7 Alternative 6: Melt and Dilute Blanket and Driver Fuel at ANL-W

Under this alternative, both the sodium-bonded blanket and driver spent nuclear fuel would be treated in the Hot Fuel Examination Facility at ANL-W using Options 2 and 3 of the melt and dilute process discussed in Section 2.3.4. Option 2 would be used for the blanket spent nuclear fuel, and Option 3 would be used for the driver spent nuclear fuel except for 0.1 metric tons (0.11 tons) of oxide, carbide, and nitride fuel, which would not be treated under the alternative. **Figure 2–28** illustrates the steps for the alternative.

Removal of the sodium from the blanket spent nuclear fuel and, to the extent practical, from the driver spent nuclear fuel would take place at the Hot Fuel Examination Facility. Equipment for sodium removal activities and the melt and dilute process would need to be installed in the inert cell of the facility.

The metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be transferred to the Radioactive Scrap and Waste Facility for storage pending ultimate disposal.

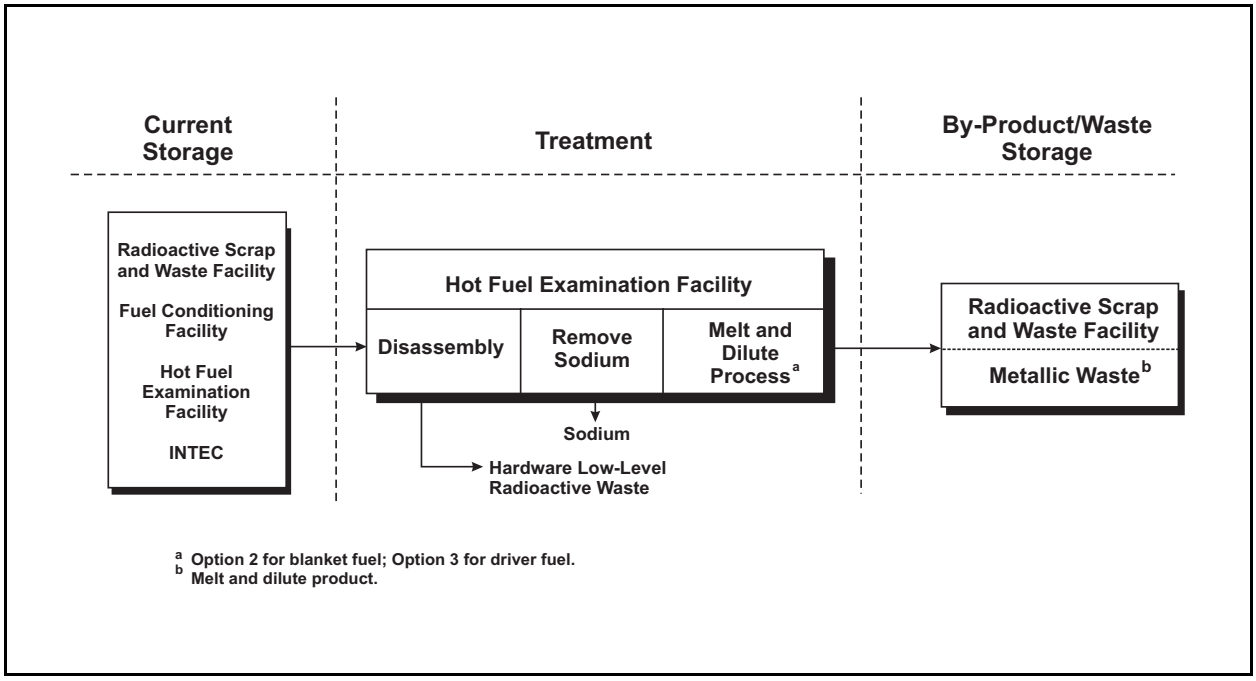

Figure 2–28 Schematic for Driver and Blanket Spent Nuclear Fuel in Alternative 6

The melt and dilute process at ANL-W could start as early as 2003 and would take approximately 12 years to complete for all driver and blanket sodium-bonded spent nuclear fuel.

2.6 ALTERNATIVES CONSIDERED AND DISMISSED

In identifying the reasonable alternatives for evaluation in this EIS, two separate issues led to the determination of alternatives that were considered and dismissed: (1) the level of maturity of the alternative technologies; and (2) the level of effort required to modify an existing facility to implement a specific technology. The construction of new facilities when existing facilities are still operative was not considered a reasonable option because of impacts and cost implications. Among the treatment technologies discussed in Section 2.3, the GMODS process and the direct plasma arc-vitreous ceramic process are not as mature as the electrometallurgical, melt and dilute, and PUREX processes when applied to sodium-bonded spent nuclear fuel. The GMODS and plasma arc processes both require significant and extensive research and development before they can be successfully proven to treat sodium-bonded spent nuclear fuel. The GMODS and plasma arc-vitreous ceramic processes each present specific technological challenges that cannot be answered without the construction, operation, and considerable engineering analysis of pilot-scale plants. In comparison, the melt and dilute process is being tested and evaluated, and has been selected as the Preferred Alternative for treatment of almost all aluminum-based spent nuclear fuel at SRS (DOE 2000). Use of the melt and dilute process for sodium-bonded driver spent nuclear fuel only requires technology enhancement, which DOE already has proposed for treating other spent nuclear fuel. In addition, unlike the other technologies that would require no new construction, GMODS and plasma arc processes would require the installation of large, specialized equipment in new hot cell facilities, the size and complexity of which are not sufficiently determined to allow detailed environmental impact analysis.

GMODS Process

The GMODS process, although similar to the melt and dilute process because of its thermal treatment, has not been developed beyond the laboratory scale. Several developmental steps would be required before it could be deemed a mature process. These include: detailed process development, resolution of containment concerns, testing, and a pilot plant demonstration to address technology risks (e.g., reliability and throughput).

GMODS would require large, specialized equipment to be installed in eight new large hot cell facilities. GMODS would dissolve the fuel elements or fuel assemblies entirely in a lead/lead-oxide system. An off-gas treatment system similar to that for the melt and dilute process would be required to treat the radioactive elements volatilized at about 1,000 °C (1,830 °F). The GMODS equipment could produce an intermediate waste form containing most of the actinides, fission products, and structural materials. After some preprocessing, the waste stream would be fed into the melter for the production of a new type of borosilicate glass log. These logs would contain uranium, other actinides, and structural elements in addition to the fission products. Because of the highly corrosive nature of the chemicals in the system, the technical feasibility of the alternative has not been established. This would add an additional degree of uncertainty to the waste estimates, as well as to the ultimate success of the fuel conditioning project.

Direct Plasma Arc-Vitreous Ceramic Process

The direct plasma arc-vitreous ceramic process is being used for the vitrification of low-level mixed waste. However, vitrification of spent nuclear fuel by this process is understood only on a conceptual level. The plasma arc treatment method would require large, complex equipment to be installed in a new, specially constructed hot cell facility. Such a facility could be constructed next to the Hot Fuel Examination Facility at ANL-W to secure some services. It would require the installation of equipment to cut the fuel assemblies into small pieces, a ceramic melter (furnace) to melt and oxidize the pieces at temperatures at least as high as 1,600 °C (2,900 °F), and an off-gas treatment system. As with the GMODS and melt and dilute processes, uranium and plutonium are not separated during the process. The conditioned spent nuclear fuel form would be vitreous ceramic and would include the sodium in a stable form. As with all processes that dissolve or melt spent nuclear fuel, the plasma arc process would produce radioactive off-gases. These gases would be filtered and treated, and the filter and treatment media would be stabilized into an acceptable waste form by a yet-to-be-determined process. The process would require testing in a pilot-scale plant to address the reliability of the plasma system.

The high temperatures of the process could increase the radioactive materials available for release during normal operation and accident conditions, thus increasing the exposure risk to members of the general public. Compared to other alternatives, there is a substantial uncertainty about the risk from accident conditions, considering the complexity of the off-gas treatment system. Because of the high temperature, more radioactive elements would be volatilized. In addition, considerable development would be required to produce very high- temperature rotating equipment that would operate reliably in a hot cell environment.

Chloride Volatility Process

The chloride volatility process design is in an early conceptual stage. The process needs high temperatures and chlorination for volatilization and chemical reactions to separate various fission products from uranium. This treatment technology would require a very elaborate gaseous separation process with potentially significant occupational and public risk in comparison to other treatment technologies, from both the volatilized fission products and the chlorine gas.

Electrometallurgical Treatment at INEEL Test Area North

Treatment of sodium-bonded spent nuclear fuel using the electrometallurgical treatment process at INEEL's Test Area North was considered and dismissed because Test Area North would require extensive modification to treat sodium-bonded spent nuclear fuel. Implementation of this alternative would require the construction of an argon hot cell. In addition, it would require either the procurement of new equipment or the transfer of already-contaminated equipment and other systems existing at ANL-W.

Treatment of Driver or Clad Blanket Spent Nuclear Fuel Using SRS PUREX Process

As discussed in Section 2.3.2, use of the PUREX process facilities at SRS for the treatment of sodium-bonded spent nuclear fuel would require the development and installation of a versatile front-end process to handle mechanical decladding, sodium removal, and zirconium sludge formation for EBR-II spent nuclear fuel. Such development does not appear justified for the sole purpose of treating the relatively small quantity of driver spent nuclear fuel.

Treatment of Driver Spent Nuclear Fuel Using SRS Melt and Dilute Process

As discussed in Section 2.3.4, the treatment of driver spent nuclear fuel would require a modified melt and dilute process that would be capable of handling the sodium volatilized from processing chopped driver spent nuclear fuel elements with the cladding intact. To accomplish this at SRS, significant design changes would be required from the process that DOE has proposed for the aluminum-clad spent nuclear fuel, which does not contain sodium. These design changes do not appear justified for the sole purpose of treating the relatively small quantity of driver spent nuclear fuel.

Treatment Using INEEL PUREX Process

Sodium-bonded spent nuclear fuel from EBR-II was being processed at the Idaho Chemical Processing Plant (now INTEC) using a PUREX process. DOE stopped processing at INTEC as a matter of policy in 1992, and the facility was permanently shut down. Reactivation of the facility is not practical and the alternative was dismissed.

2.7 ULTIMATE DISPOSITION

One of the technical risks in implementing any of the sodium-bonded spent nuclear fuel treatment methods is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for placement in a potential repository. DOE would receive a license from the NRC to receive and store spent nuclear fuel in a repository (10 CFR 60 or draft 10 CFR 63). In order to obtain a license, DOE must develop acceptance criteria that establish the condition of the spent nuclear fuel for disposal and demonstrate that the criteria will meet NRC standards. Any spent nuclear fuel packaging or treatment technology must be capable of putting fuel in a form that will satisfy the acceptance criteria requirements. DOE's Office of Civilian Radioactive Waste Management has responsibility for a Federal repository. It is working to refine its acceptance criteria to ensure that spent nuclear fuel and high-level radioactive waste are packaged suitably for disposal. DOE has drafted preliminary acceptance criteria that are being used to assess the feasibility of DOE spent nuclear fuel disposition options (DOE 1999a). If the repository is developed, final acceptance criteria will not be available until after the NRC issues its construction authorization, based on the successful demonstration of safe, long-term performance of the candidate repository in accordance with NRC regulations. Until such time, the preliminary acceptance criteria tend to be conservative to allow for uncertainties in performance of engineered and natural barriers and how such performance will impact public and worker health and safety, and material isolation.

To ensure that the treatment option DOE selects will result in a product that is likely to meet the acceptance criteria, DOE is working with the NRC to obtain comments on the research and development work that DOE will perform to establish treatment technology specifications. To provide additional independent evaluation of the suitability of new treatment technologies, DOE requested that the National Academy of Sciences' National Research Council provide recommendations regarding DOE's sodium-bonded spent nuclear fuel treatment and disposition program. In a recent report (NAS 1998), the National Research Council recommended that the Electrometallurgical Treatment Research and Demonstration Project be carried to completion. The Council also expressed the opinion that, with the exception of the PUREX process, all other alternatives to the electrometallurgical process were at an early stage of development.

2.8 PREFERRED ALTERNATIVE

Council on Environmental Quality regulations (40 CFR 1502.14e) require that an agency identify its preferred alternative(s) in the final EIS. After careful consideration of public comments and programmatic, environmental, nonproliferation, and cost issues, DOE has identified electrometallurgical treatment as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel. Thus, the Preferred Alternative is a combination of Alternative 1 and the No Action Alternative. This combination would result in 26 metric tons of heavy metal of EBR-II and miscellaneous spent nuclear fuel being treated using the electrometallurgical process and 34 metric tons of heavy metal of Fermi-1 blanket spent nuclear fuel remaining in storage, pending a subsequent decision on its long-term management. The environmental consequences of the Preferred Alternative are addressed in Section 4.10.

DOE will validate the cost of using alternative treatment techniques (e.g., sodium removal and placement in high-integrity cans) for the Fermi-1 blanket spent nuclear fuel. These techniques may be economically favorable for the Fermi-1 blanket spent nuclear fuel because of characteristics that distinguish it from the EBR-II spent nuclear fuel. For example, the Fermi-1 blanket spent nuclear fuel does not require the extensive safeguards and security measures that are required for the EBR-II blanket fuel. The difference in security requirements for these two types of fuel is a result of the difference in plutonium content. The EBR-II blanket fuel has 30 times more plutonium at a greater concentration than the Fermi-1 blanket fuel.

Should DOE select the Preferred Alternative in the Record of Decision, DOE would proceed with the electrometallurgical treatment of the EBR-II sodium-bonded spent nuclear fuel and monitor the results and costs while continuing the development of sodium removal techniques for the Fermi-1 blanket spent nuclear fuel. Sodium removal would increase the number of long-term management options for the Fermi-1 fuel. While EBR-II spent nuclear fuel is undergoing electrometallurgical treatment and the Fermi-1 spent nuclear fuel remains in storage, DOE has approximately four years in which to evaluate the operating experience of electrometallurgical treatment technology and further develop other alternatives for the Fermi-1 spent nuclear fuel. After these data are evaluated, DOE would decide whether to treat the Fermi-1 blanket spent nuclear fuel using electrometallurgical treatment or to use another treatment method and/or disposal technique.

Before making a decision to treat or dispose of the Fermi-1 blanket spent nuclear fuel, DOE will determine whether the analysis in this EIS is adequate to support a subsequent Record of Decision or whether additional NEPA review is required. In any case, DOE will notify the public of its preferred approach for the Fermi-1 blanket spent nuclear fuel at least 30 days before issuing a Record of Decision regarding treatment or disposal.

For several years, DOE has been actively developing electrometallurgical treatment technology specifically for the management of sodium-bonded spent nuclear fuel. Having completed a successful demonstration of electrometallurgical treatment, DOE believes that this technology has the highest probability of meeting the Department's needs for managing much of the sodium-bonded spent nuclear fuel. Electrometallurgical

technology would convert the reactive fuel into ceramic and metallic waste forms, both of which are more stable than untreated sodium-bonded spent nuclear fuel. In addition, uranium would be separated from the spent nuclear fuel, blended with depleted uranium if needed to lower enrichment levels, and cast into ingots to be stored until a disposition decision is made through a separate NEPA review. Most of the plutonium would be disposed of in the ceramic waste form, with the remaining small fraction disposed of in the metallic waste form. Currently, the only waste form that has been tested and analyzed extensively under geologic repository conditions and may be accepted for repository disposal is borosilicate glass. Tests have shown the ceramic and metallic waste forms from electrometallurgical treatment may perform as well as the standard borosilicate glass waste form. The ceramic and metallic waste forms would require less storage volume than untreated spent nuclear fuel.

2.9 SUMMARY OF ENVIRONMENTAL IMPACTS

This section summarizes the environmental impacts associated with the No Action Alternative and the six reasonable alternatives under the proposed action that are evaluated in detail as part of this EIS (see Section 2.5). The information presented in this section is based on Chapter 4, which provides a detailed discussion of the impacts on potentially affected environmental areas. Such environmental areas include: air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, waste management, and transportation.

For the alternatives evaluated, the analyses showed that there would be no significant impacts on air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, and transportation. The radiological and nonradiological gas and liquid releases, as well as the associated exposures to workers and the public, would be well below regulatory standards and guidelines and no mitigation measures would be warranted. In addition, the environmental impact analysis indicates that there are no significant impacts that would discriminate one alternative over another.

The only significant difference between the No Action Alternative and the reasonable alternatives is in the area of waste generation. All of the proposed alternatives result in a decrease in high-level radioactive waste volume as compared to the direct disposal volume associated with the No Action Alternative.

For the No Action Alternative and the six alternatives evaluated, the proposed facilities already exist. Except for internal building modifications and new equipment installation, no construction activities would be required. Therefore, DOE has determined the proposed action would have minimal or no impacts on land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources. Impacts to these resources were not evaluated in detail in the EIS.

The impacts of the No Action Alternative are presented first as a baseline for comparing the impacts under the proposed action. A summary of the environmental impacts for the No Action Alternative and the other six reasonable alternatives is presented as **Table 2–4**. Section 4.1, Chapter 4, provides information on the categories and results illustrated in Table 2–4.

2.9.1 No Action Alternative Impacts

Under the No Action Alternative, the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed from the interior of the fuel elements). The EIS evaluates the impacts of two separate options under this alternative:

a. Monitoring and stabilizing the sodium-bonded spent nuclear fuel as necessary for continued storage at current locations until 2035, or until a new treatment technology (such as GMODS or plasma arc) is developed.

b. Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository by packaging the fuel in high-integrity cans. As discussed in Section 2.3.8, the direct disposal option may not meet current NRC and/or RCRA requirements.

The activities associated with the preparation of sodium-bonded spent nuclear fuel for direct disposal would be similar to those needed to prepare the fuel for interim or continued storage. Both require that fuel be transferred to a hot cell, examined (nondestructive examination) and characterized, and repackaged. The only difference between these two options is that for direct disposal, the sodium-bonded spent nuclear fuel would be placed in high-integrity cans in preparation for ultimate disposal, while for storage it would not be placed in high-integrity cans. Direct disposal also requires consideration of criticality safety, thereby limiting the amount of driver spent nuclear fuel that could be packaged in a canister, leading to higher repository volume requirements.

Air Quality

For both options under the No Action Alternative, activities at ANL-W and INTEC would have a negligible impact on existing air quality. Radiological emissions would also be low and well below regulatory concern. Approximately 810 curies would be released over a 35-year period from possible fuel degradation during storage. Air quality for INEEL is not expected to change as a result of the No Action Alternative.

Water Resources

Surface water is not used at ANL-W and INTEC and this would not change under either option of the No Action Alternative. Groundwater use, primarily domestic consumption, could decrease if there is a reduction in workers at ANL-W.

No changes are expected in liquid effluent discharges. There are currently no discharges to surface waters (radiological or nonradiological) except for discharges of nonhazardous liquid waste, which are monitored and subject to National Pollutant Discharge Elimination System (NPDES) permit requirements.

Socioeconomics

Under either option of the No Action Alternative, there could be a reduction of approximately 350 workers at ANL-W. This reduction could result in the loss of 940 additional indirect jobs in the economic region. The reduction would take place over time, therefore, the No Action Alternative would not result in any noticeable changes in the existing regional economy, housing characteristics, or community services.

Public and Occupational Health and Safety

The risk to the health and safety of workers and the public under either option of the No Action Alternative would be from the potential exposure to radiological or hazardous chemical emissions during normal operation or accident conditions.

Radiological Exposures

Routine radioactive releases associated with either option of the No Action Alternative at ANL-W and INTEC would be small. The maximum annual dose to the population within 80 kilometers (50 miles) from these releases would be 0.0015 person-rem. This exposure would lead to 7.5×10^{-7} additional latent cancer fatalities in the population, or one chance in 1.3 million that the exposed population would experience a latent cancer fatality. For comparison purposes, the collective dose for the same population in the year 2010 from natural background radiation would be 86,250 person-rem. The maximally exposed offsite individual

would receive a maximum of 0.00026 millirem, and the risk of developing a fatal cancer from this exposure would be 1.3×10^{-10}, or one chance in 7.7 billion. The average worker would receive 60 millirem per year, and the risk of developing a cancer from this exposure would be 0.000024, or one chance in 41,666.

Similarly, for the 35-year duration of the project, the population dose from routine releases would be 0.013 person-rem (with a latent cancer fatality risk of 6.5×10^{-6}); the maximally exposed offsite individual dose would be 0.0023 millirem (with a latent cancer fatality risk of 1.1×10^{-9}); and the worker population dose would be 209 person-rem (with a latent cancer fatality risk of 0.084).

The maximum annual cancer risk from postulated accident conditions under continued storage or direct disposal of the No Action Alternative at ANL-W would be 5.6×10^{-6} for the population within 80 kilometers (50 miles). The annual cancer risk for the maximally exposed offsite individual would be 4.8×10^{-8}, and 1.5×10^{-8} for the noninvolved worker.

Hazardous Chemical Exposures

Hazardous chemical impacts resulting from either option of the No Action Alternative would be small because any emissions of hazardous chemicals from activities under the No Action Alternative would be very low.

Hazardous chemical impacts under accident conditions, evaluated in terms of Emergency Response Planning Guideline (ERPG) values, indicate that under either option of the No Action Alternative, the worst postulated accident conditions would result in less than ERPG-1 conditions for a worker or the maximally exposed offsite individual.

Environmental Justice

As discussed above, the impacts from either option of the No Action Alternative on the health and safety of the public would be very small regardless of the racial and ethnic composition of the population and independent of the economic status of the individuals comprising the population in 2010.

Waste Management

For both options under the No Action Alternative, various types of waste would continue to be generated at ANL-W. These include low-level radioactive, transuranic, mixed, hazardous, and nonhazardous waste. These waste types are associated with the operation of the facilities where the sodium-bonded spent nuclear fuel is stored. High-level radioactive waste in metallic and ceramic forms generated as a result of completing the Electrometallurgical Treatment Research and Demonstration Project waste processing would be stored at the Radioactive Scrap and Waste Facility pending disposal. Finally, some additional low-level radioactive waste and transuranic waste would be generated from the deactivation of the demonstration project. The volumes of these waste types are presented in Table 2–4.

Transportation

The No Action Alternative involves the movement of sodium-bonded spent nuclear fuel within the INEEL site. All fuel stored at INTEC would be moved to ANL-W for repackaging in preparation for transport either to a geologic repository or out of the State of Idaho by 2035.

The dose to transportation workers from all transportation activities under the No Action Alternative is estimated at 0.003 person-rem; the dose to the public would be 0.022 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.2×10^{-6} latent cancer fatalities among transportation

workers and 0.000011 latent cancer fatalities in the total affected population over the duration of the transportation activities.

The dose to the affected population from postulated accidents during transportation would be less than 1.0×10^{-6} person-rem, resulting in less than 1.0×10^{-9} latent cancer fatalities. Nonradiological transportation accidents are estimated to result in 0.00012 traffic fatalities.

2.9.2 Proposed Action Impacts

Under the proposed action, the EIS evaluates six distinct alternatives, as described in Section 2.5 and illustrated in Figure 2-22. Alternative 1 proposes to treat both driver and blanket spent nuclear fuel using the electrometallurgical method at ANL-W. Alternatives 2 through 5 propose to treat the driver spent nuclear fuel using the electrometallurgical method (as in Alternative 1), but other methods and/or sites would be used for the blanket spent nuclear fuel, including: the high-integrity can packaging at ANL-W (Alternative 2); the PUREX process at SRS (Alternative 3); the melt and dilute process at ANL-W (Alternative 4); and the melt and dilute process at SRS (Alternative 5). Alternative 6 proposes to treat both driver and blanket spent nuclear fuel using the melt and dilute method at ANL-W.

All alternatives under the proposed action would have very small impacts on air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, and transportation areas of the environment in and around the INEEL/ANL-W and SRS locations. For all alternatives, the radiological and nonradiological gaseous emissions and liquid effluent, as well as the associated exposures to workers and the public, would be well below regulatory standards and guidelines. A major difference between the No Action and proposed action alternatives is in the area of waste generation. Since the acceptability of chemically reactive sodium in a high-level radioactive waste repository is a primary concern in this EIS, it is important to consider the volume of high-level radioactive waste for all the proposed action alternatives. All the proposed action alternatives would result in a decrease in high-level radioactive waste volume as compared to the direct disposal No Action Alternative. The reduction in high-level radioactive waste volume for each alternative would be: 47 percent (Alternative 1); 71 percent (Alternative 2); 84 percent (Alternative 3); 58 percent (Alternative 4); 37 percent (Alternative 5); and 43 percent (Alternative 6).

Air Quality

The proposed action would have a negligible impact on existing air quality at ANL-W and SRS for each of the alternatives. Air quality at ANL-W and SRS would not change as a result of the proposed action.

During the year of maximum releases, radiological gaseous emissions would be well below regulatory concerns for each of the alternatives. Radiological gaseous emissions at ANL-W would be in the range of 770 (Alternative 1) to 2,162 (Alternative 6) curies of elemental tritium and 11,600 (Alternative 1) to 32,650 (Alternative 6) curies of krypton-85.

During the year of maximum releases, radiological gaseous emissions at SRS would be 54 (Alternative 5) to 162 (Alternative 3) curies of elemental tritium and 396 (Alternative 5) to 1,187 (Alternative 3) curies of krypton-85.

Water Resources

Surface water is not used at ANL-W, and this would not change under any of the alternatives proposed for ANL-W. Groundwater use, primarily domestic consumption, would remain at current levels, as the work force would be expected to remain at current levels for all alternatives of the proposed action.

No changes would be expected in liquid effluent discharges from any of the alternatives at ANL-W. There are currently no discharges to surface waters (radiological or nonradiological) except for discharges of nonhazardous liquid waste to the industrial pond, which are monitored and are subject to NPDES permit requirements.

Potential radioactive liquid effluent has been identified for the PUREX process at SRS under Alternative 3. Table 2–4 indicates some small quantities of tritium, and other isotopes. No radioactive liquid effluent has been identified for the melt and dilute process at SRS under Alternative 5.

Socioeconomics

All the alternatives under the proposed action assume that the treatment and management of the sodium-bonded spent nuclear fuel at ANL-W or SRS would not require an additional work force, but the activities would keep the work force from being reduced. Therefore, there would be no changes to the socioeconomic conditions in the vicinity of either ANL-W or SRS.

Public and Occupational Health and Safety

The potential risk of concern to the health and safety of the workers and the public under the proposed action would be from exposure to routine radiological emissions and hazardous chemical releases under normal operation or accident conditions. As indicated in Table 2–4, the risk would be small for all alternatives considered under the proposed action.

Radiological Exposures

Comparing alternatives at ANL-W, the maximum annual population dose from routine gaseous radioactive releases would range from 0.0028 person-rem (Alternative 1) to 0.012 person-rem (Alternative 6), with latent cancer fatalities in the range of 1.4×10^{-6} to 6.0×10^{-6}, respectively. The project total population dose would range from 0.016 person-rem (Alternative 1) to 0.024 person-rem (Alternative 6), with latent cancer fatalities in the range of 8.2×10^{-6} to 1.2×10^{-5}, respectively.

The maximum annual dose to the maximally exposed offsite individual at ANL-W would range from 0.00034 millirem (Alternative 1) to 0.002 millirem (Alternative 6), with a latent cancer fatality risk in the range of 1.7×10^{-10} to 1.0×10^{-9}, respectively. The project total dose to the maximally exposed offsite individual would range from 0.002 millirem (Alternative 1) to 0.004 (Alternative 6), with latent cancer fatality risks of 1.0×10^{-9} to 2.0×10^{-9}, respectively.

The collective annual dose to workers at ANL-W would be 22 person-rem for all alternatives. This corresponds to additional latent cancer fatalities of 0.0088. The average dose to a worker at ANL-W would be 60 millirem per year, which corresponds to a latent cancer fatality risk of 0.000024 per year. The project total worker dose would range from 231 person-rem (Alternatives 3 and 5) to 319 person-rem (Alternatives 1 and 4), with latent cancer fatalities ranging between 0.092 and 0.13, respectively.

Comparing alternatives at SRS, the maximum population dose from routine gaseous radioactive releases would range from 0.0076 person-rem per year for three years (Alternative 5) to 0.02 person-rem for the whole treatment period (Alternative 3), corresponding to additional latent cancer fatalities in the range of 0.000011 to 0.00001, respectively, for the whole treatment period.

The maximum dose to the maximally exposed offsite individual would range from 0.0001 millirem per year for three years (Alternative 5) to 0.00051 millirem (Alternative 3) for the whole treatment period, with latent cancer fatality risks of 1.5×10^{-10} and 2.6×10^{-10}, respectively, for the whole treatment period.

The maximum collective dose to workers at SRS would be 50 person-rem per year (Alternative 5) for three years. This corresponds to 0.06 additional latent cancer fatalities. The maximum average dose to a worker at SRS would be 500 millirem per year (Alternative 5), which corresponds to a latent cancer fatality risk of 0.0002 per year.

The highest annual latent cancer fatality risk for the population within 80 kilometers (50 miles) of ANL-W from postulated design-basis accident conditions under the proposed action would be 0.009 (Alternative 6, driver fuel, design-basis earthquake). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 0.000076 (Alternative 6, driver fuel, design-basis earthquake). The highest annual latent cancer fatality risk for the noninvolved worker would be 2.7×10^{-6} (Alternative 6, driver fuel, design-basis earthquake).

The highest annual latent cancer fatality risk for the population within 80 kilometers (50 miles) of ANL-W from postulated beyond-design-basis accident conditions under the proposed action would be 0.000013 (Alternatives 1 through 5, driver fuel, beyond-design-basis earthquake). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 2.2×10^{-7} (Alternatives 1 through 5, driver fuel, beyond-design-basis earthquake). The highest annual latent cancer fatality risk for the noninvolved worker would be 2.3×10^{-9} (Alternatives 1 through 5, blanket fuel, beyond-design-basis earthquake).

The highest annual latent cancer fatality risk for the population within 80 kilometers (50 miles) of SRS from postulated design-basis accident conditions under the proposed action would be 0.011 (Alternative 5, blanket fuel, loss of power). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 6.6×10^{-6} (Alternative 5, blanket fuel, loss of power). The highest annual latent cancer fatality risk for the noninvolved worker would be 3.4×10^{-7} (Alternative 5, blanket fuel, loss of power).

Hazardous Chemical Exposures

Hazardous chemical impacts from normal operations for all alternatives under the proposed action would be small because the emissions of hazardous chemicals from the treatment and management of sodium-bonded spent nuclear fuel would be very low.

Hazardous chemical impacts under accident conditions, evaluated in terms of comparison to ERPG values, indicate that under the proposed action, all postulated hazardous chemical releases would not result in worse than ERPG-1 conditions for the noninvolved worker or the maximally exposed offsite individual at either ANL-W or SRS.

Environmental Justice

As discussed above, the impacts from the proposed action on the health and safety of the public would be very small, regardless of the racial and ethnic composition of the population and independent of the economic status of the individuals comprising the population in 2010.

Waste Management

Table 2–4 presents a comparison of the volumes of high-level radioactive, low-level radioactive, and transuranic waste generated by each of the alternatives. The alternatives would generate from 37 to 84 percent less high-level radioactive waste as compared to the direct disposal option of the No Action Alternative. Each of the alternatives would generate more transuranic waste, but Alternatives 1, 2, 4, and 6 would exceed this waste volume by a range of only 7 to 41 percent. Alternatives 3 and 5 would generate significantly greater volumes of transuranic waste, between 2.7 to 10 times the volume of transuranic waste generated by the direct disposal No Action Alternative.

All of the alternatives either would remove or convert the metallic sodium into a nonreactive form.

With respect to disposability and waste acceptance criteria, only the borosilicate glass waste form of Alternative 3 for blanket spent nuclear fuel has been tested and analyzed extensively under conditions relevant to a geologic repository. It is expected that other waste forms (e.g., ceramic, metallic, and possibly high-integrity cans not containing metallic sodium) would be suitable for repository disposal.

Transportation

The transportation activities for all alternatives under the proposed action would involve the movement of the sodium-bonded spent nuclear fuel within the INEEL site.

The incident-free dose to transportation workers from these activities would be between 0.0043 (Alternative 2) and 0.027 (Alternative 6) person-rem; the dose to the affected public would be between 0.032 (Alternative 2) and 0.2 (Alternative 6) person-rem. Accordingly, incident-free transportation activities would result in latent cancer fatalities in the range of 1.7×10^{-6} to 1.1×10^{-5} among transportation workers and 0.000016 to 0.0001 among the total affected population over the duration of the transportation activities.

The dose to the affected population from postulated accidents from these activities would be less than 1×10^{-6} person-rem, resulting in less than 1×10^{-9} latent cancer fatalities. Nonradiological traffic fatalities would be approximately 0.0001.

Transportation activities under Alternatives 3 and 5 would include, in addition, the movement of the blanket spent nuclear fuel pins from ANL-W to SRS. The incident-free dose to transportation workers from these activities would be 0.0012 person-rem; the dose to the public would be 0.012 person-rem. Accordingly, incident-free transportation activities would result in 4.7×10^{-7} latent cancer fatalities among transportation workers and 6.0×10^{-6} latent cancer fatalities in the total affected population over the duration of the transportation activities. Nonradiological fatalities from vehicle emissions during intersite transportation would be 0.00039 among affected urban populations along the transportation route.

The dose to the affected population from postulated accidents from these activities would be less than 3.0×10^{-6} person-rem, resulting in less than 1.5×10^{-9} additional latent cancer fatalities. Nonradiological traffic fatalities would be 0.0018.

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Table 2–4 Summary of Environmental Consequences for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Resource/Material Categories	*No Action*		*Alternative 1*		*Alternative 2*	
	ANL-W		**ANL-W**		**ANL-W**	
Air Quality - Radiological air emissions (curies per year)	Negligible impact 811 [a]		Negligible impact Tritium: 770 Krypton-85: 11,600		Negligible impact Tritium: 809 Krypton-85: 11,860	
Water Resources - Radiological liquid effluent (curies per year)	No impact No liquid effluent		No impact No liquid effluent		No impact No liquid effluent	
Socioeconomics	Loss of 350 direct jobs and 940 indirect jobs. No noticeable impact.		Work force maintained; No impact		Work force maintained; No impact	
Public and Occupational Health and Safety						
• Project duration	35 years		13 years		9 years	
• Normal operations (annual) [c]	**Dose per year**	**LCF**	**Dose per year**	**LCF**	**Dose per year**	**LCF**
- Population	0.0015	7.5×10^{-7}	0.0028	1.4×10^{-6}	0.003	1.5×10^{-6}
- MEI	0.00026	1.3×10^{-10}	0.00034	1.7×10^{-10}	0.00037	1.9×10^{-10}
- Average individual	6.2×10^{-6}	3.1×10^{-12}	0.000012	5.8×10^{-12}	0.000013	6.2×10^{-12}
- Worker population	22	0.0088	22	0.0088	22	0.0088
- Average worker	60	0.000024	60	0.000024	60	0.000024
• Normal operations (project total) [c]	**Dose**	**LCF**	**Dose**	**LCF**	**Dose**	**LCF**
- Population	0.013	6.5×10^{-6}	0.016	8.2×10^{-6}	0.017	8.3×10^{-6}
- MEI	0.0023	1.1×10^{-9}	0.002	1.0×10^{-9}	0.0021	1.0×10^{-9}
- Worker population	209	0.084	319	0.13	231	0.092
Hazardous chemicals						
- MEI	None		None		None	
• Accidents Maximum annual cancer risk						
- Population	5.6×10^{-6} (DBA) [a]		5.6×10^{-6} (DBA); 0.000013 (BDBA)		5.6×10^{-6} (DBA); 0.000013 (BDBA)	
- MEI	4.8×10^{-8} (DBA) [a]		4.8×10^{-8} (DBA); 2.2×10^{-7} (BDBA)		4.8×10^{-8} (DBA); 2.2×10^{-7} (BDBA)	
- Noninvolved worker	1.5×10^{-8} (DBA) [a]		4.5×10^{-8} (DBA); 2.3×10^{-9} (BDBA)		1.5×10^{-8} (DBA); 1.5×10^{-9} (BDBA)	
Chemical accidents						
- MEI	Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
- Noninvolved worker	Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
Environmental Justice	No disproportionately high and adverse impact to minority or low-income populations					
Waste Management (cubic meters)						
• High-level radioactive waste	152 (Direct disposal volume) [d]		81.1		43.2 [e]	
• Low-level radioactive waste	904		862		733.7	
• Transuranic waste	12.1		14.1		10.7	
Transportation						
• Incident-free	**Person-rem**	**LCF**	**Person-rem**	**LCF**	**Person-rem**	**LCF**
- Population	0.022	0.000011	0.033	0.000016	0.032	0.000016
- Workers	0.003	1.2×10^{-6}	0.0044	1.8×10^{-6}	0.0043	1.7×10^{-6}
• Accidents						
- Population	less than 1.0×10^{-6}	less than 1.0×10^{-9}	less than 1.0×10^{-6}	less than 1.0×10^{-9}	less than 1.0×10^{-6}	less than 1.0×10^{-9}

ERPG = Emergency Response Planning Guideline; LCF = Latent Cancer Fatalities; MEI = Maximally Exposed Offsite Individual;
DBA = Design-Basis Accident; BDBA = Beyond-Design-Basis Accident

[a] Represents total curies for 35 years; tritium: 51 curies; krypton-85: 760 curies; iodine-129: 0.000018 curies.

[b] Over a period of six months.

Alternative 3				*Alternative 4*		*Alternative 5*				*Alternative 6*	
ANL-W		**SRS** [b]		**ANL-W**		**ANL-W**		**SRS**		**ANL-W**	
Negligible impact Tritium: 809 Krypton-85: 11,860		Negligible impact Tritium: 162 Krypton-85: 1,187		Negligible impact Tritium: 809 Krypton-85: 11,860		Negligible impact Tritium: 809 Krypton-85: 11,860		Negligible impact Tritium: 54 Krypton-85: 396		Negligible impact Tritium: 2,162 Krypton-85: 32,650	
No impact No liquid effluent		Negligible impact Tritium: 1.54 Other: less than 0.022		No impact No liquid effluent		No impact No liquid effluent		No impact No liquid effluent		No impact No liquid effluent	
Work force maintained; no impact		Work force maintained; no impact		Work force maintained; no impact		Work force maintained; no impact		Work force maintained; no impact		Work force maintained; no impact	
9 years		Less than 1 year		13 years		9 years		3 years		12 years	
Dose per year	**LCF**	**Dose per year**	**LCF**	**Dose per year**	**LCF**	**Dose per year**	**LCF**	**Dose per year**	**LCF**	**Dose per year**	**LCF**
0.003	1.5×10^{-6}	0.02	0.000010	0.003	1.5×10^{-6}	0.003	1.5×10^{-6}	0.0076	3.8×10^{-6}	0.012	6.1×10^{-6}
0.00037	1.9×10^{-10}	0.00051	2.6×10^{-10}	0.00037	1.9×10^{-10}	0.00037	1.9×10^{-10}	0.00010	5.0×10^{-11}	0.002	1.0×10^{-9}
0.000013	6.2×10^{-12}	0.000024	1.2×10^{-11}	0.000013	6.2×10^{-12}	0.000013	6.2×10^{-12}	0.000011	5.5×10^{-12}	0.000051	2.6×10^{-11}
22	0.0088	38	0.015	22	0.0088	22	0.0088	50	0.02	22	0.0088
60	0.000024	250	0.0001	60	0.000024	60	0.000024	500	0.0002	60	0.000024
Dose	**LCF**	**Dose**	**LCF**	**Dose**	**LCF**	**Dose**	**LCF**	**Dose**	**LCF**	**Dose**	**LCF**
0.017	8.3×10^{-6}	0.02	0.00001	0.017	8.3×10^{-6}	0.017	8.3×10^{-6}	0.023	0.000011	0.024	0.000012
0.0021	1.0×10^{-9}	0.00051	2.6×10^{-10}	0.0021	1.0×10^{-9}	0.0021	1.0×10^{-9}	0.0003	1.5×10^{-10}	0.004	2.0×10^{-9}
231	0.092	38	0.015	319	0.13	231	0.092	150	0.06	297	0.12
None		None		None		None		None		None	
5.6×10^{-6} (DBA); 0.000013 (BDBA)		0.00014 (DBA)		0.00022 (DBA); 0.000013 (BDBA)		5.6×10^{-6} (DBA); 0.000013 (BDBA)		0.011		0.0090 (DBA)	
4.8×10^{-8} (DBA); 2.2×10^{-7} (BDBA)		7.2×10^{-8} (DBA)		1.9×10^{-6} (DBA); 2.2×10^{-7} (BDBA)		4.8×10^{-8} (DBA); 2.2×10^{-7} (BDBA)		6.6×10^{-6}		0.000076 (DBA)	
1.5×10^{-8} (DBA); 1.5×10^{-9} (BDBA)		6.2×10^{-7} (DBA)		4.9×10^{-8} (DBA); 1.5×10^{-9} (BDBA)		1.5×10^{-8} (DBA); 1.5×10^{-9} (BDBA)		3.4×10^{-7}		2.7×10^{-6} (DBA)	
Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
No disproportionately high and adverse impacts to minority or low-income populations											
23.6 (18 at ANL-W; 5.6 at SRS)				63.6		94.62 (18 at ANL-W; 76.62 at SRS)				86	
2,960.5 (770.5 at ANL-W; 2,190 at SRS)				845		1,178.5 (770.5 at ANL-W; 408 at SRS)				924	
100.7 (10.7 at ANL-W; 90 at SRS)				12.8		27.2 (10.7 at ANL-W; 16.5 at SRS)				14.1	
Person-rem	**LCF**	**Person-rem**	**LCF**	**Person-rem**	**LCF**	**Person-rem**	**LCF**	**Person-rem**	**LCF**	**Person-rem**	**LCF**
0.03	0.000015	0.0012	6.0×10^{-6}	0.14	0.000072	0.03	0.000015	0.0012	6.0×10^{-6}	0.2	0.0001
0.004	1.6×10^{-6}	0.0012	4.7×10^{-7}	0.02	7.9×10^{-6}	0.004	1.6×10^{-6}	0.0012	4.7×10^{-7}	0.027	0.000011
less than 1.0×10^{-6}	less than 1.0×10^{-9}	less than 3.0×10^{-6}	less than 1.5×10^{-9}	less than 1.0×10^{-6}	less than 1.0×10^{-9}	less than 1.0×10^{-6}	less than 1.0×10^{-9}	less than 3.0×10^{-6}	less than 1.5×10^{-9}	less than 1.0×10^{-6}	less than 1.0×10^{-9}

[c] Annual dose represents the maximum dose in a single year. Population doses (population and worker population) are in person-rem; individual doses are in millirem. The regulatory dose limit for offsite individuals (public) is 10 millirem per year from air exposures and 100 millirem per year for all pathways. The administrative control limit for an individual worker at a DOE site is 2,000 millirem per year.

[d] Includes 142 cubic meters of spent nuclear fuel.

[e] Includes 25.2 cubic meters of spent nuclear fuel.

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3. AFFECTED ENVIRONMENT

Chapter 3 provides an overview of the affected environment of the alternative sites under consideration for the treatment and management of sodium-bonded spent nuclear fuel. The chapter first addresses the approach to defining the affected environment, and then provides a discussion of the affected environment at the Idaho National Engineering and Environmental Laboratory and the Savannah River Site. The discussion of each resource area at each site initially addresses the site as a whole, followed by a description of the proposed treatment locations.

3.1 APPROACH TO DEFINING THE AFFECTED ENVIRONMENT

In accordance with the Council on Environmental Quality guidance under National Environmental Policy Act (NEPA) regulations (40 CFR 1500 through 1508) for preparing an environmental impact statement (EIS), the affected environment is "interpreted comprehensively to include the natural and physical environment and the relationship of people with that environment." The affected environment descriptions presented in this chapter provide the context for understanding the environmental consequences described in Chapter 4. They serve as a baseline for identifying and evaluating the environmental changes that may result from implementing any of the alternatives.

Candidate sites for the treatment and management of sodium-bonded spent nuclear fuel include the U.S. Department of Energy's (DOE) Argonne National Laboratory-West (ANL-W), located within the boundaries of the Idaho National Engineering and Environmental Laboratory (INEEL), and the Savannah River Site's (SRS) F-Area and L-Area. The affected environment is described for the following resource areas: land use, site infrastructure, air quality and noise, water resources, geology and soils, ecological resources, cultural and paleontological resources, socioeconomics, environmental justice, existing human health risk, and waste management. For each DOE site, each resource area is described first for the site as a whole and then for the candidate treatment sites, as appropriate. The level of detail varies depending on the potential for impacts resulting from each treatment and management alternative.

The affected environment for each candidate site presented in this section is based on the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999h), unless otherwise noted. Additional information on the affected environment was determined from other recent EISs, previous environmental studies, relevant laws and regulations, and other government reports and databases. More detailed information on the affected environment at the candidate sites can be found in annual site environmental reports and site NEPA documents such as the *Idaho National Engineering and Environmental Laboratory Advanced Mixed Waste Treatment Project Final Environmental Impact Statement* (DOE 1999a) and the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000).

3.2 IDAHO NATIONAL ENGINEERING AND ENVIRONMENTAL LABORATORY

INEEL is located on approximately 230,700 hectares (570,000 acres) in southeastern Idaho and is 55 kilometers (34 miles) west of Idaho Falls; 61 kilometers (38 miles) northwest of Blackfoot; and 35 kilometers (22 miles) east of Arco. INEEL is owned by the Federal Government and administered, managed, and controlled by DOE. It is primarily within Butte County, but portions of the site are also in Bingham, Jefferson, Bonneville, and Clark counties. The site is roughly equidistant from Salt Lake City, Utah, and Boise, Idaho.

There are 450 buildings and 2,000 support structures at INEEL, with more than 279,000 square meters (3,000,000 square feet) of floor space in varying conditions of utility. INEEL has approximately 25,100 square meters (270,000 square feet) of covered warehouse space and an additional 18,600 square meters (200,000 square feet) of fenced yard space. The total area of the various machine shops is 3,035 square meters (32,665 square feet).

Fifty-two research and test reactors have been used at INEEL over the years to test reactor systems, fuel and target design, and overall safety. In addition to nuclear research reactors, other INEEL facilities are operated to support reactor operations. These facilities include high- and low-level radioactive waste processing and storage sites; hot cells; analytical laboratories; machine shops; and laundry, railroad, and administrative facilities. Other activities include management of one of DOE's largest storage sites for low-level radioactive waste and transuranic waste.

3.2.1 Land Resources

3.2.1.1 Land Use

The Federal Government, the State of Idaho, and private parties own lands surrounding INEEL. Regional land uses include grazing, wildlife management, rangeland, mineral and energy production, recreation, and crop production. Approximately 60 percent of the surrounding area is used by sheep and cattle for grazing. Small communities and towns near the INEEL boundaries include Mud Lake to the east; Arco, Butte City, and Howe to the west; and Atomic City to the south. Two national natural landmarks border INEEL: Big Southern Butte (2.4 kilometers [1.5 miles] south) and Hell's Half Acre (2.6 kilometers [1.6 miles] southeast). A portion of Hell's Half Acre National Natural Landmark is designated as a Wilderness Study Area. The Black Canyon Wilderness Study Area also is adjacent to the northwest boundary of INEEL.

Land-use categories at INEEL include facility operations, grazing, general open space, and infrastructure (e.g., roads). Generalized land uses at INEEL and within the vicinity are shown in **Figure 3–1.** Facility operations include industrial and support operations associated with energy research and waste management activities. Up to 340,000 acres (137,600 hectares) of the site is leased for cattle and sheep grazing; grazing permits are administered by the Bureau of Land Management (DOE 1999i). Land also is used for recreation and environmental research associated with the designation of INEEL as a National Environmental Research Park. Much of INEEL is open space that has not been designated for specific use. Some of this space serves as a buffer zone between INEEL facilities and other land uses. Recently, approximately 29,950 hectares (74,000 acres) of open space in the north central portion of the site have been designated as the INEEL Sagebrush Steppe Ecosystem Reserve (DOE 1999g). This area represents some of the last sagebrush steppe ecosystem in the United States and provides a home for a number of rare and sensitive species of plants and animals. About 2 percent of the total INEEL site area (4,600 hectares [11,400 acres]) is used for facilities and operation. INEEL facilities are sited within a central core area of about 93,100 hectares (230,000 acres) (Figure 3-1). Public access to most facilities is restricted. Approximately 6 percent of INEEL (34,000 acres [13,760 hectares]) is devoted to utility rights-of-way and public roads (DOE 1999i). DOE land-use plans and policies applicable to INEEL are discussed in the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (Programmatic Spent Nuclear Fuel EIS) (DOE 1995a).

The total land area at ANL-W is 328 hectares (810 acres); however, site facilities cover only about 20 hectares (50 acres), or 6 percent of the site (DOE 1996a). ANL-W is located 7 kilometers (4.3 miles) northwest of the nearest site boundary and is designated as a testing center for advanced technologies associated with nuclear power systems. The area has 52 major buildings, including reactor buildings, laboratories, warehouses, technical and administrative support buildings, and craft shops that comprise 55,700 square meters (600,000 square feet) of floor space (LMITCO 1997). Five nuclear test reactors, including the Experimental

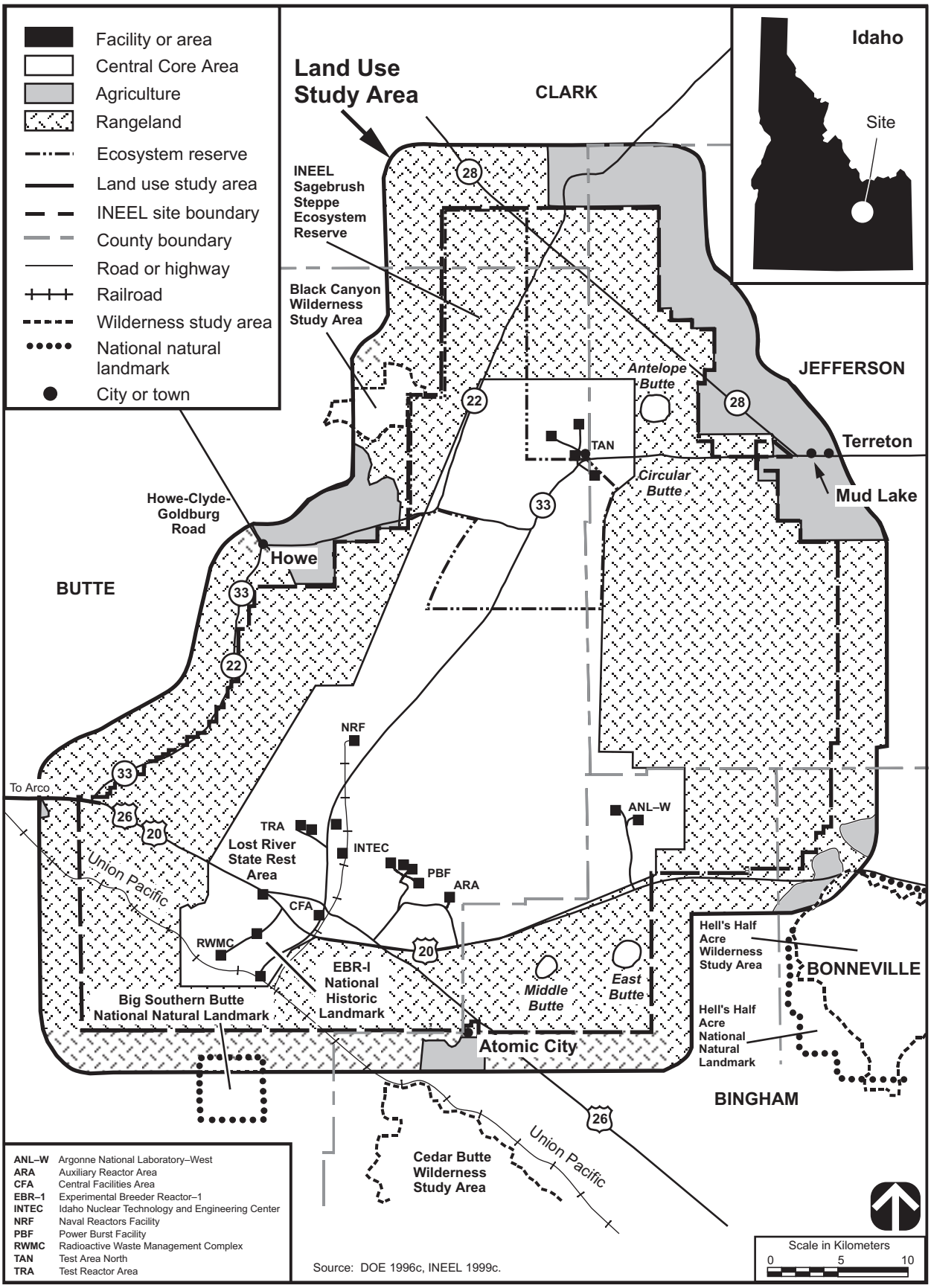

Figure 3–1 Generalized Land Use at INEEL and Vicinity

Breeder Reactor II (EBR-II), have operated on the site, although the only one currently active is a small reactor used for radiography examination of experiments, waste containers, and spent nuclear fuel. The Fuel Conditioning Facility and the Hot Fuel Examination facility also are located at the site (DOE 1996a).

3.2.1.2 Visual Resources

The Bitterroot, Lemhi, and Lost River mountain ranges border the INEEL site on the north and west. Volcanic buttes near the southern boundary of INEEL can be seen from most locations on the site. Lands adjacent to the site under Bureau of Land Management jurisdiction are designated as Visual Resource Management Class II areas. However, the Black Canyon Wilderness Study Area, located adjacent to the northwest site boundary, is under consideration by the Bureau of Land Management for Wilderness designation. If approved, this area would upgrade its Visual Resource Management rating to Class I. INEEL itself generally consists of open desert land mostly covered by big sagebrush and grasslands. Most of the land within the site falls within Visual Resource Management Class II and III. Management activities within these classes may be seen but should not dominate the view (DOI 1986).

Ten facility areas are located on the INEEL site. Although INEEL has a comprehensive facility and land use plan (LMITCO 1997), no specific visual resource standards have been established. INEEL facilities appear as low-density commercial/industrial complexes widely dispersed throughout the site. Structure heights range from about 3 to 30 meters (10 to 100 feet); a few stacks and towers reach 76 meters (250 feet). Although many INEEL facilities are visible from highways, most facilities are more than 0.8 kilometers (0.5 miles) from public roads. The operational areas are well defined at night by security lights.

Developed areas within ANL-W are consistent with a Visual Resource Management Class IV rating in which management activities dominate the view and are the focus of viewer attention. The tallest structure at ANL-W is the Fuel Conditioning Facility stack, which is 61 meters (200 feet) in height. The site is visible from Highway 20. Facilities that stand out from the highway include the Hot Fuel Examination Facility, the EBR-II containment shell, the Zero Power Physics Reactor, and the Transient Reactor Test Facility. Natural features of visual interest within a 40-kilometer (25-mile) radius of ANL-W include the East Butte at 9 kilometers (5.6 miles), Middle Butte at 11 kilometers (6.8 miles), Hell's Half Acre National Natural Landmark and Hell's Half Acre Wilderness Study Area at 15 kilometers (9.3 miles), Big Lost River at 19 kilometers (11.8 miles), and Big Southern Butte National Natural Landmark at 30 kilometers (18.6 miles).

3.2.2 Site Infrastructure

Site infrastructure includes those utilities and other resources required to support modification and continued operation of mission-related facilities identified under the various alternative actions. INEEL has extensive production, service, and research facilities. An extensive infrastructure system supports these facilities, as shown in **Table 3–1**.

3.2.2.1 Transportation

The road network at INEEL provides for onsite transportation; railroads are used for deliveries of large volumes of coal and oversized structural components. Commercial shipments are transported by truck; some bulk materials are transported by train; and waste is transported by truck and train. About 140 kilometers (87 miles) of paved surface have been developed out of the 445 kilometers (277 miles) of roads on the site, including 29 kilometers (18 miles) of service roads that are closed to the public. Most of the roads are adequate for the current level of normal transportation activity and could handle increased traffic volume.

Idaho Falls receives railroad freight service from Butte, Montana, to the north, and from Pocatello, Idaho, and Salt Lake City, Utah, to the south. The Union Pacific Railroad's Blackfoot-to-Arco Branch crosses the

southern portion of INEEL and provides rail service to the site. This branch connects with a DOE spur line at Scoville Siding, then links with developed areas within INEEL. There are 48 kilometers (30 miles) of railroad track at INEEL. Rail shipments to and from INEEL usually are limited to bulk commodities, spent nuclear fuel, and radioactive waste.

Table 3–1 INEEL Site-Wide Infrastructure Characteristics

Resource	*Current Usage*	*Site Capacity*
Transportation		
Roads (kilometers)	445 [a]	Not applicable
Railroads (kilometers)	48	Not applicable
Electricity		
Energy consumption (megawatt hours per year)	221,772 [b]	394,200
Peak load (megawatts)	39 [b]	124
Fuel		
Natural gas (cubic meters per year)	Not applicable	Not applicable
Oil and propane (liters per year)	5,820,000	16,000,000 [c]
Coal (metric tons per year)	11,340	11,340 [c]
Water (liters per year)	4,900,000,000 [d]	43,000,000,000 [e]

[a] Includes paved and unpaved roads.
[b] Fiscal Year 1997 data based on INEEL 1998.
[c] Low supplies can be replenished by truck or rail.
[d] 1997 usage based on DOE 1999a.
[e] Water right allocation.
Source: DOE 1999h, except as noted in footnotes b and d.

3.2.2.2 Electricity

Commercial electric power is supplied to INEEL through two feeders from the Antelope substation to the federally owned Scoville substation, which supplies electric power directly to the site's electric power distribution system. Electric power supplied by Idaho Power Company is generated by hydroelectric generators along the Snake River in southern Idaho and by the Bridger and Valmy coal-fired thermal electric generation plants in southwestern Wyoming and northern Nevada.

The average electrical availability at INEEL is about 394,200 megawatt hours per year; in 1997 the average usage was 221,772 megawatt hours. The peak load capacity for INEEL is 124 megawatts; the 1997 peak load usage was about 39 megawatts (INEEL 1998). Current electrical usage at ANL-W is 28,700 megawatt hours per year (Goff 1999).

3.2.2.3 Fuel

Fuel consumed at INEEL include several types of liquid petroleum fuel, coal, and propane gas. All fuel is transported to the site for use and storage. Fuel storage is provided for each facility, and the inventories are restocked as necessary. The current site usage of fuel oil is about 5.8 million liters (1.5 million gallons) per year. The current site usage of coal is about 11,340 metric tons (12,500 tons) per year. If additional coal or fuel oil were needed during the year, it could be shipped to the site.

3.2.2.4 Water

The Snake River Plain Aquifer is the source of all water at INEEL. The water is provided by a system of about 30 wells, together with pumps and storage tanks. That system is administered by DOE, which holds

the Federal Reserved Water Right of 43 billion liters (11.4 billion gallons) per year for the site. The current site usage is about 4.9 billion liters (1.3 billion gallons) per year (DOE 1999a).

3.2.2.5 Site Safety Services

DOE operates three fire stations at INEEL. These stations are at the north end of Test Area North, at ANL-W, and in the Central Facilities Area. Each station has a minimum of one engine company capable of supporting any fire emergency in its assigned area. The fire department also provides the site with ambulance, emergency medical technician, and hazardous material response services.

3.2.3 Air Quality and Noise

3.2.3.1 Air Quality

The climate at INEEL and the surrounding region is characterized as a semiarid steppe with low relative humidity, wide daily temperature swings, and large variations in annual precipitation. The average annual temperature at INEEL is 5.6 °C (42 °F), and average seasonal temperatures range from a minimum of -7.3 °C (18.8 °F) in winter to 18.2 °C (64.8 °F) in summer. Temperature extremes range from a summertime maximum of 39.4 °C (103 °F) to a wintertime minimum of -45 °C (-49 °F). The average annual precipitation at INEEL is 22 centimeters (8.7 inches). Prevailing winds at INEEL are predominantly southwest or northeast, although terrain features may cause variations in the flow (DOE 1999a). The average annual wind speed is 3.4 meters per second (7.5 miles per hour).

Nonradiological Releases

INEEL is within Eastern Idaho Intrastate Air Quality Control Region No. 61. None of the areas within INEEL or its surrounding counties are designated as nonattainment areas, i.e., areas where criteria air pollutant levels exceed the National Ambient Air Quality Standards (NAAQS) established by the U.S. Environmental Protection Agency (EPA) (40 CFR 50). The nearest nonattainment area for particulate matter is in Pocatello, about 80 kilometers (50 miles) to the south. Applicable NAAQS and Idaho State ambient air quality standards are presented in **Table 3–2**.

The primary sources of nonradiological air pollutants at INEEL currently include calcination of sodium-bearing waste, combustion of coal for steam, and combustion of fuel oil for heating. Other emission sources include waste burning, industrial processes, stationary diesel engines, vehicles, and fugitive dust from activities including waste burial and construction. The existing ambient air concentrations attributable to sources at INEEL are presented in Table 3–2. For criteria pollutants, concentrations are based on (1) dispersion modeling at the INEEL site boundary centered at the Idaho Nuclear Technology and Engineering Center (INTEC) facility, performed for the High-Level Waste and Facilities Disposition EIS (DOE 1999i) using 1997 actual emissions and excluding ANL-W; and (2) dispersion modeling at the INEEL site boundary centered on ANL-W using 1997 actual emissions. The modeling performed for the High-Level Waste and Facilities Disposition EIS used EPA's ISCST3 model with hourly meteorological data. The ANL-W modeling used EPA's SCREEN3 model, which is very conservative compared to ISCST3, and uses a set of worst-case meteorological conditions to predict a maximum one-hour concentration. This one-hour concentration was converted to other averaging times using regulatory scaling factors (SCDHEC 1993). For these reasons, the ANL-W concentrations are extremely conservative. In spite of this conservatism, total maximum site boundary concentrations (which can be approximated by summing individual concentrations) are well below ambient air quality standards.

For acrolein, cadmium, toluene, and xylene, the concentrations in Table 3–2 are based on dispersion modeling using actual INEEL site-wide emissions for the year 1997 and modeling these emissions as if they

all originated from ANL-W. For the remaining hazardous/toxic air pollutants, concentrations are based on site-wide dispersion modeling using maximum emissions for the year 1990, as presented in the Programmatic Spent Nuclear Fuel EIS (DOE 1995a). Only those hazardous/toxic air pollutants that would be emitted for any of the alternatives evaluated in this EIS are presented. The hazardous/toxic air pollutant standards are presented for informational purposes only, as the standards apply only to new or modified emissions sources.

Table 3–2 Comparison of Modeled Ambient Air Concentrations at the INEEL Site Boundary From INEEL Sources With Most Stringent Applicable Standards or Guidelines

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*	*Maximum INEEL Concentration Excluding ANL-W for Criteria Pollutants* [a] *(micrograms per cubic meter)*	*Maximum ANL-W Concentration* [b] *(micrograms per cubic meter)*
Criteria Pollutants [a]				
Carbon monoxide	8 hours 1 hour	10,000 [c] 40,000 [c]	78 206	41 59
Nitrogen dioxide	Annual	100 [c]	0.46	13
Ozone	8 hours	157 [d]	(e)	(e)
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 [c] 150 [c] 150 [d]	0.49 12 (f)	0.14 1.1 (f)
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 [d] 65 [d]	(f) (f)	(f) (f)
Sulfur dioxide	Annual 24 hours 3 hours	80 [c] 365 [c] 1,300 [c]	0.14 5.3 24	3.3 27 60
Hazardous/Toxic Air Pollutants [g]				
1,3-Butadiene	Annual	0.0036	0.001	Not applicable
Acetaldehyde	Annual	0.45	0.0110	Not applicable
Acrolein [h]	24 hours	12.5	0.00332	Not applicable
Benzene	Annual	0.12	0.0290	Not applicable
Cadmium [h]	Annual	0.00056	0.0000415	Not applicable
Formaldehyde	Annual	0.077	0.012	Not applicable
Toluene [h]	24 hours	18,750	0.392	Not applicable
Xylene [h]	24 hours	21,750	0.0362	Not applicable

PM_n = Particulate matter less than or equal to *n* microns in diameter.

[a] Concentrations for criteria pollutants from the High-Level Waste and Facilities Disposition EIS, based on actual emissions plus projected increases, for dispersion modeling centered on INTEC with no contribution from ANL-W (DOE 1999i: Table C 2-14, No Action Alternative). Concentrations for hazardous and toxic compounds were estimated based on site-wide modeling of 1990 emissions, as presented in the Programmatic Spent Nuclear Fuel EIS.

[b] Concentrations for criteria pollutants based on dispersion modeling centered on ANL-W, using 1997 ANL-W actual emissions.

[c] Federal and state standards.

[d] Standard currently under litigation, but will become enforceable during the life of the project.

[e] Not directly emitted or monitored by the site.

[f] No data are available with which to assess particulate matter concentrations.

[g] Note that standards apply only to new or modified sources and are provided for informational purposes only.

[h] Estimated based on 1997 INEEL emissions, modeling all emissions as if they originated from ANL-W.

Sources: 40 CFR 50, Rules 577, 585, 586; ID APA 16.01.01, DOE 1995a, DOE 1999i, 62 FR 38855, 62 FR 38652.

The nearest Prevention of Significant Deterioration Class I area[1] to INEEL is Craters of the Moon Wilderness Area, Idaho, located 53 kilometers (33 miles) west-southwest from the center of the site. There are no other Class I areas within 100 kilometers (62 miles) of INEEL. INEEL and its vicinity are classified as a Prevention of Significant Deterioration Class II area[2].

The EPA has established Prevention of Significant Deterioration increments for certain pollutants: sulfur dioxide, nitrogen dioxide, and particulate matter less than or equal to 10 microns in diameter (PM_{10}). The increments specify a maximum allowable increase above a certain baseline concentration for a given averaging period, and apply only to sources constructed or modified after a specified baseline date. These sources are known as increment-consuming sources. The baseline date is the date of submittal of the first application for a Prevention of Significant Deterioration permit in a given area.

Prevention of Significant Deterioration permits have been obtained for the coal-fired, steam-generating facility (located next to INTEC) and the Fuel Processing Facility. Operation of the Fuel Processing Facility is not expected (DOE 1996c). In addition to these facilities, INEEL has other increment-consuming sources on site. **Tables 3–3** and **3–4** specify the current amount of Prevention of Significant Deterioration increment consumption in Class I and Class II areas, respectively, by INEEL's increment-consuming sources based on dispersion modeling analyses.

Table 3–3 Prevention of Significant Deterioration Increment Consumption at Craters of the Moon Wilderness (Class I) Area by Existing (1996) and Projected Sources Subject to Prevention of Significant Deterioration Regulation[a]

Pollutant	*Averaging Time*	*Allowable Prevention of Significant Deterioration Increment* [b] *(micrograms per cubic meter)*	*Amount of Prevention of Significant Deterioration Increment Consumed (micrograms per cubic meter)*
Nitrogen dioxide [c]	Annual	2.5	0.004
Respirable particulates [d]	Annual	4	0.008
	24 hours	8	0.6
Sulfur dioxide	Annual	2	0.09
	24 hours	5	1.8
	3 hours	25	5.9

[a] Projected sources include emissions between the present and the time that the proposed Advanced Mixed Waste Treatment Facility becomes operational.

[b] All increments specified are State of Idaho standards (ID DHW Rule 581; ID APA 16.01.01).

[c] Assumes that the New Waste Calcining Facility (the largest source of nitrogen dioxide emissions at INEEL) operates for the entire year.

[d] Data on particulate size are not available for most sources. For purposes of comparison to the respirable particulate increments, it is assumed conservatively that all particulates emitted are of respirable size (i.e., 10 microns or less in diameter).

Source: DOE 1999a.

Routine offsite monitoring for nonradiological air pollutants generally is performed only for particulates. Monitoring for PM_{10} is performed by the Environmental Science and Research Foundation at the site boundary and at communities beyond the boundary. In 1997, 49 samples were collected at Rexburg (located about 65 kilometers [40 miles] east of the site). The mean PM_{10} concentration at Rexburg was 14 micrograms per cubic meter. Forty-one samples were collected at the Mountain View Middle School in Blackfoot in 1997, with a mean concentration of 15 micrograms per cubic meter. Twenty-nine samples were collected at Atomic City in 1997, with a mean concentration of 15 micrograms per cubic meter (Evans et al. 1998).

[1] *Class I areas are defined as national parks and wildlife refuges.*

[2] *Class II areas are defined as any area not designated Class I. Please see Appendix B, Impact Assessment Methods, for a more detailed discussion.*

Table 3–4 Prevention of Significant Deterioration Increment Consumption at Class II Areas by Existing (1996) and Projected Sources Subject to Prevention of Significant Deterioration Regulation at INEEL [a]

Pollutant	*Averaging Time*	*Allowable Prevention of Significant Deterioration Increment [b] (micrograms per cubic meter)*	*Amount of Prevention of Significant Deterioration Increment Consumed [c] (micrograms per cubic meter)*
Nitrogen dioxide [d]	Annual	25	1.3
Respirable particulates [e]	Annual	17	0.1
	24 hours	30	3.8
Sulfur dioxide	Annual	20	1.8
	24 hours	91	12
	3 hours	512	74

[a] Projected sources include emissions between the present and the time that the proposed Advanced Mixed Waste Treatment Facility becomes operational.
[b] All increments specified are State of Idaho standards (ID DHW Rule 581; ID APA 16.01.01).
[c] Maximum concentration predicted at the INEEL site boundary.
[d] Assumes that the New Waste Calcining Facility operates for the entire year.
[e] Data on particulate size are not available for most sources. For purposes of comparison to the respirable particulate increments, it is assumed conservatively that all particulates emitted are of respirable size (i.e., 10 microns or less in diameter).
Source: DOE 1999a.

Some monitoring data also have been collected by the National Park Service at the Craters of the Moon Wilderness Area. The monitoring program has shown no cases in which the primary 1-hour ozone and total suspended particulate standards and low levels of sulfur dioxide were exceeded (except for one case in which the 24-hour standard was exceeded in 1985) (DOE 1999a). Note that the total suspended particulates within standards have been replaced with PM_{10} standards and the 1-hour ozone standard has been replaced by the 8-hour standard.

The primary sources of nonradiological air emissions at ANL-W include four water tube boilers for site heating and process requirements, various emergency or standby diesel generators used for backup power, a permitted paint spray booth, a permitted decontamination facility at the Fuel Conditioning Facility, and two fixed-roof storage tanks that hold fuel for the boilers (DOE 1998a).

Radiological Releases

The primary sources of radiological air pollutants at INEEL as well as the localized releases at ANL-W are presented in **Table 3–5**. During 1997, an estimated 5,596 curies of radioactivity were released to the atmosphere from all INEEL sources. Ninety-five percent of the total airborne radioactive effluent was released from two INEEL facilities, the Test Reactor Area and ANL-W. The Test Reactor Area released a total of 1,695 curies, of which over 93 percent was from radioisotopes of noble gases. ANL-W released 3,605 curies, of which radioisotopes of noble gases comprised over 99 percent.

Year-to-year fluctuations in airborne radioactive effluent releases are dependent on which processes are active at INEEL facilities. The total for 1997 is considerably higher than the annual totals for 1993 to 1996, due primarily to the 3,579 curies of krypton-85 released from ANL-W. Krypton-85, a noble gas, was released from ANL-W as part of a spent fuel treatment project, the Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West. Although these 1997 releases are higher than previous years, they are still considerably less than the annual totals for the 1980s.

Table 3–5 Radiological Gaseous and Airborne Emissions at INEEL in 1997 (Curies)

Emission Type	*Radionuclide* [a]	*ANL-W*	*Other Facilities at INEEL* [b]	*Total* [c]
Noble gases	Argon-41	3.9	1,550	1,554
	Krypton-85	3,579	—	3,579
	Xenon-135	—	20.9	20.9
	Krypton-88	—	3.5	3.5
	Krypton-85m	—	3.0	3.0
	Xenon-133	—	3.4	3.4
	Krypton-87	—	1.8	1.8
	Xenon-138	—	0.8	0.8
	Xenon-135m	—	1.1	1.1
Airborne particulates	Rubidium-88	—	1.3	1.3
	Rubidium-89	—	0.011	0.011
	Cesium-138	—	0.069	0.069
	Chromium-51	—	0.0056	0.0056
	Sodium-24	—	0.014	0.0056
	Cesium-137	—	0.0071	0.0071
	Technetium-99m	—	0.0022	0.0022
	Antimony-125	—	0.000027	0.000027
	Strontium-90 [d]	—	0.00070	0.00070
	Plutonium-238	—	5.1×10^{-6}	5.1×10^{-6}
	Plutonium-239	1.1×10^{-7}	1.5×10^{-6}	1.6×10^{-6}
Tritium, C-14, and iodine isotopes	Tritium (H-3)	23.0	403	426
	Carbon-14	—	0.91	0.91
	Iodine-129	—	0.058	0.058
	Iodine-131	—	0.0017	0.0017
	Iodine-133	—	0.00055	0.00055
Others		0.000039	0.0035	0.0035
Totals		**3,606**	**1,990**	**5,596**

[a] The table includes all radionuclides with total releases greater than 10^{-7} curies. Values are not corrected for decay after release.
[b] Facilities include INTEC, the Test Reactor Area, Naval Reactor Facility, Central Facility Area, Radioactive Waste Management Complex, and Power Burst Facility.
[c] Rounded totals include small amounts from facilities not listed.
[d] Parent-daughter equilibrium assumed.
Source: Evans et al. 1998.

3.2.3.2 Noise

Major noise emission sources within INEEL include various industrial facilities, equipment, and machines. Most INEEL industrial facilities are far enough from the site boundary that noise levels at the boundary would not be measurable or would be barely distinguishable from background levels.

Existing INEEL-related noises of public significance are from the transportation of people and materials to and from the site and in-town facilities via buses, trucks, private vehicles, helicopters, and freight trains. Noise measurements recorded 15 meters (50 feet) from U.S. Route 20 indicate that the sound levels from traffic range from 64 to 86 decibels A-weighted, and that the primary source is buses (71 to 80 decibels A-weighted). While

few people reside within 15 meters (50 feet) of the roadway, the results indicate that INEEL traffic noise might be objectionable to members of the public residing near principal highways or busy bus routes. Noise levels along these routes may have decreased somewhat because of reductions in employment and bus service at INEEL in the last few years. The acoustic environment along the INEEL site boundary in rural areas and at nearby areas away from traffic noise is typical of a rural location: the day-night average sound level is in the range of 35 to 50 decibels A-weighted (DOE 1999h). The noise generated at INEEL is not propagated at detectable levels off site, since all public areas are at least 2.5 kilometers (4 miles) away from site facilities.

No distinguishing noise characteristics at ANL-W have been identified. ANL-W is 9 kilometers (5.6 miles) from the site boundary; thus, the contributions from the area to noise levels at the site boundary are not measurable.

3.2.4 Water Resources

3.2.4.1 Surface Water

Three intermittent streams drain the mountains near INEEL: Big Lost River, Little Lost River, and Birch Creek (**Figure 3–2**). These intermittent streams carry snowmelt in the spring and are usually dry by midsummer. Several years can pass before any offsite waters enter DOE property. Big Lost River and Birch Creek are the only streams that regularly flow onto the INEEL site. Little Lost River is usually dry by the time it reaches the site because of upstream use of the flow for irrigation. None of the streams flow from the site to offsite areas. The Big Lost River discharges into the Big Lost River sinks, and there is no surface discharge from these sinks (Barghusen and Feit 1995, DOE 1996c).

The Big Lost River has been classified by the State of Idaho for domestic and agricultural use, cold water biota development, salmon spawning, primary and secondary recreation, and other special resource uses. Surface waters, however, are not used for drinking water on the site, nor is effluent discharged directly to them. Since INEEL facilities currently do not discharge directly to nor make withdrawals from these water bodies, there are no surface water rights issues at INEEL. None of the rivers have been classified as a Wild and Scenic River (DOE 1995a, DOE 1996c).

A preliminary study of the 100-year peak flow for the Big Lost River has been completed by the U.S. Geological Survey (USGS 1998). Additional studies of the 100-year and 500-year flood plains were conducted by the Bureau of Reclamation for DOE (DOE 1999i).

There are no named streams within the ANL-W area and no permanent, natural, surface water features near the area (ANL 1998a). Neither the 100-year flood nor flooding scenarios that involve the failure of Mackay Dam on the Big Lost River indicate that flood waters would reach ANL-W (Koslow and Van Haaften 1986, USGS 1998, DOE 1999i) (**Figure 3–3**).

Nonradiological Releases

ANL-W discharges 11,900,000 liters (3,140,000 gallons) per year of nonhazardous liquid waste to the sewage pond and 68,000,000 liters (18,000,000 gallons) per year to the industrial waste pond (ANL 1999b). These are evaporation ponds and water levels may be controlled by land spreading if necessary (Cascade Earth Sciences 1998).

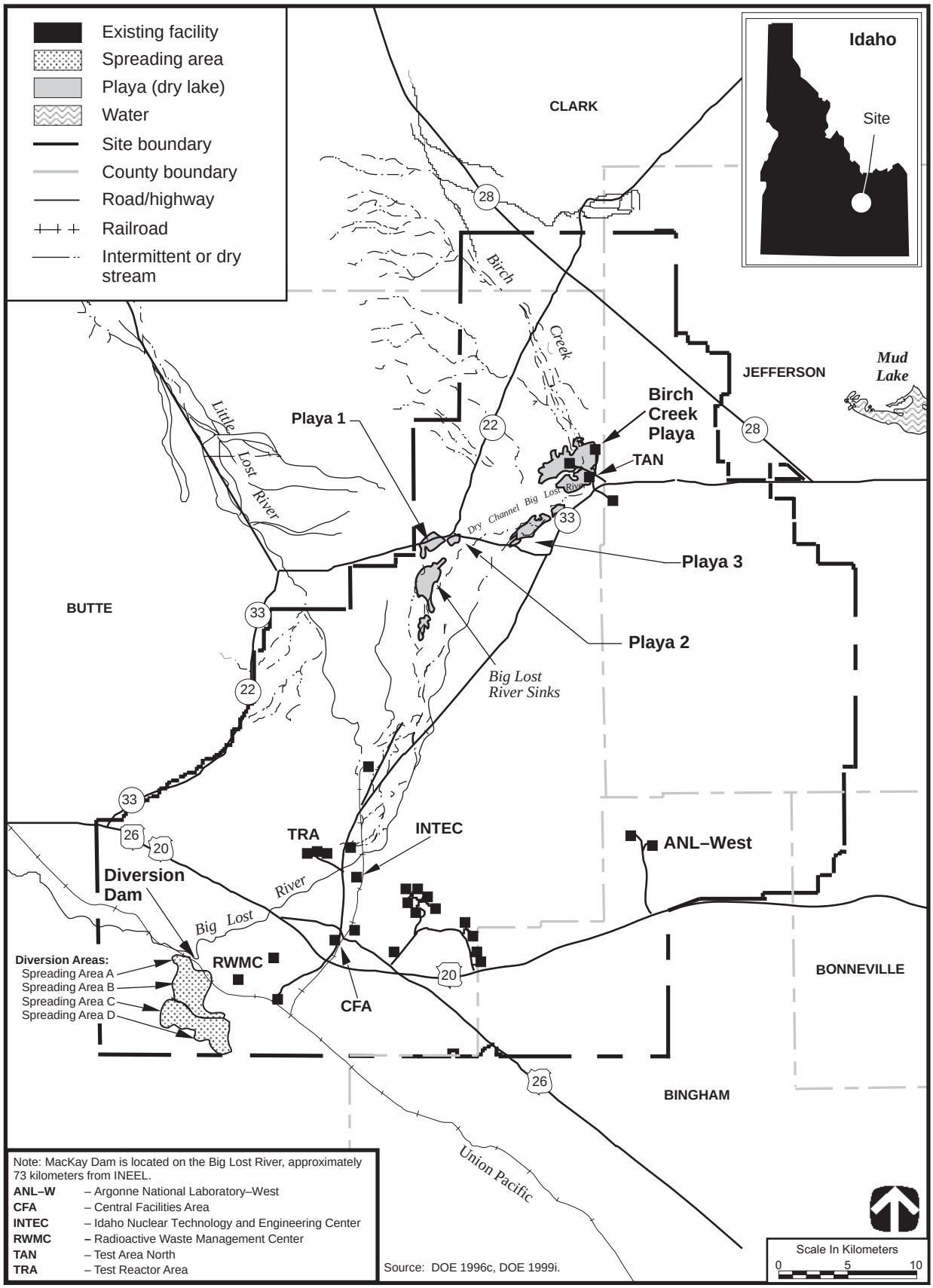

Figure 3–2 Surface Water Features at INEEL

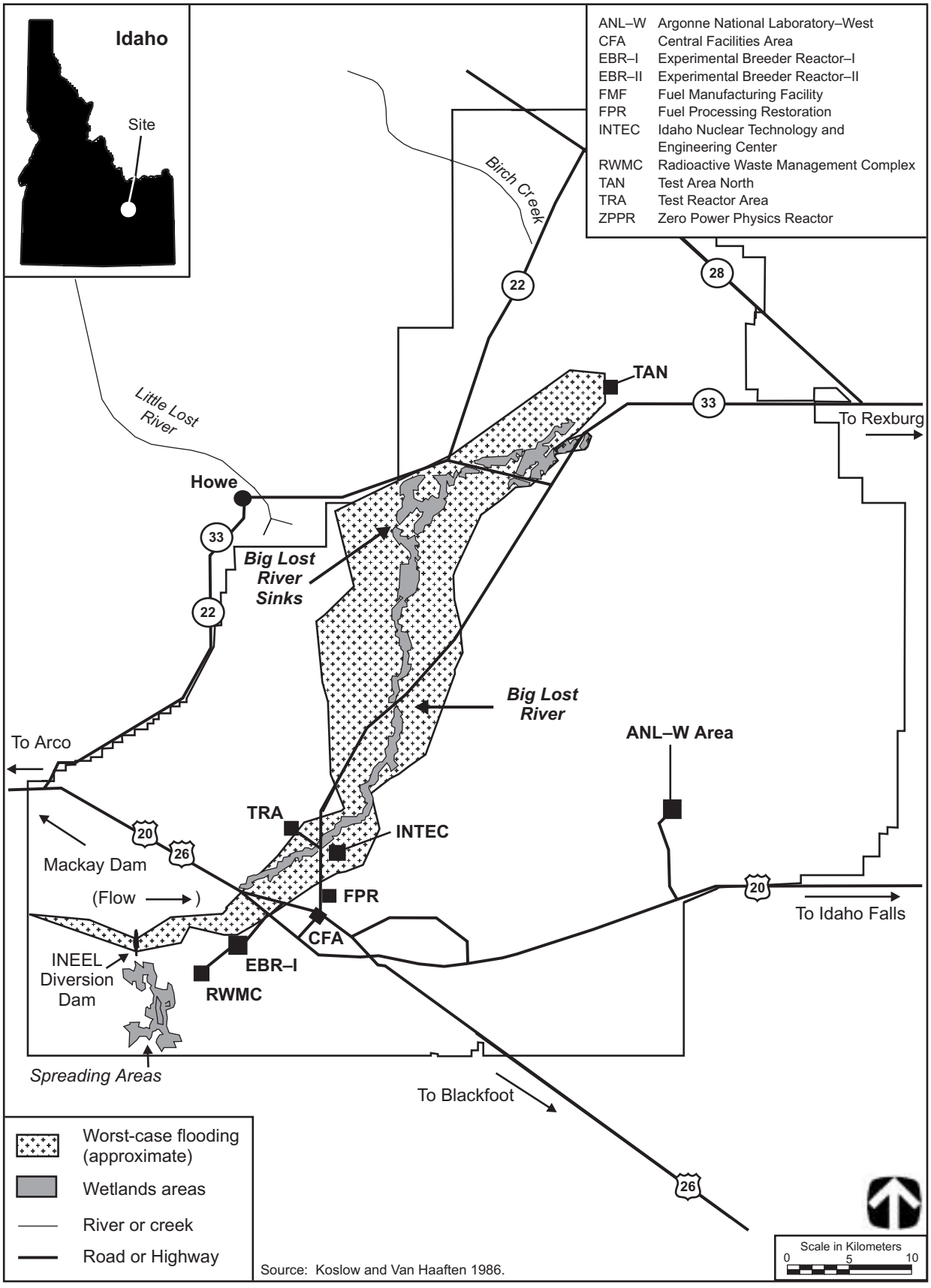

Figure 3–3 Flood Area for the Probable Maximum Flood-Induced Overtopping Failure of the Mackay Dam

Radiological Releases

Table 3–6 summarizes the radioactive liquid effluent released on site during 1997. Virtually all of the 1997 radioactive liquid effluent was released from the Test Reactor Area into two hypalon plastic-lined evaporation ponds which have been in use since August 1993. These ponds serve to prevent contaminant percolation into the ground, thus minimizing contaminant dispersal. No radioactive liquid effluent was released to the offsite environment from INEEL facilities during 1997. Routine injections of radioactive liquid effluent into the Snake River Plain Aquifer ceased in 1984 (Evans et al. 1998).

Table 3–6 Radiological Liquid Effluent Released at INEEL in 1997 (Curies)

Radionuclide [a]	*ANL-W*	*Other Facilities at INEEL* [b]	*Total*
Tritium (H-3)	—	96.3	96.3
Chromium-51	—	2.4	2.4
Cobalt-60	—	0.4	0.4
Hafnium-181	—	0.081	0.081
Strontium-90	—	0.031	0.031
Cesium-137	—	0.017	0.017
Plutonium-239	—	0.0035	0.0035
Gross Beta [c]	—	0.50	0.50
All others	—	0.06	0.06
Total	—	**99.8**	**99.8**

[a] The table includes all radionuclides with total releases greater than 0.001 curies. Values are not corrected for decay after release.
[b] Facilities include INTEC and the Test Reactor Area.
[c] Gross beta assumed to be radioactive strontium.
Source: Evans et al. 1998.

3.2.4.2 Groundwater

Aquifers are classified by Federal and state authorities according to use and quality. The Federal classifications include Class I, II, and III groundwater. Class I groundwater is either the sole source of drinking water or is ecologically vital. Class IIA and IIB are current or potential sources of drinking water (or other beneficial use), respectively. Class III is not considered a potential source of drinking water and is of limited beneficial use.

The Snake River Plain aquifer is classified by the EPA as a Class I sole source aquifer. It lies below the INEEL site and covers about 2,486,000 hectares (6,143,000 acres) in southeastern Idaho. This aquifer serves as the primary drinking water source in the Snake River Basin and is believed to contain 1.2 quadrillion to 2.5 quadrillion liters (317 trillion to 660 trillion gallons) of water. Recharge of the groundwater comes from Henry's Fork of the Snake River, Big Lost River, Little Lost River, and Birch Creek. Rainfall and snowmelt also contribute to the aquifer's recharge.

Groundwater generally flows laterally at a rate of 1.5 to 6.1 meters (5 to 20 feet) per day. It emerges in springs along the Snake River from Milner to Bliss, Idaho (DOE 1996c). Depth to the groundwater table ranges from about 61 meters (200 feet) below ground in the northeast corner of INEEL to about 275 meters (900 feet) in the southeast corner (DOE 1999i). Perched water tables (i.e., bodies of groundwater lying above a more extensive aquifer) occur below the surface. These perched water tables tend to slow the migration of pollutants that might otherwise reach the Snake River Plain aquifer.

INEEL has a large network of monitoring wells—about 120 in the Snake River Plain aquifer and another 100 drilled in the perched zone. The wells are used for monitoring to determine the compliance of specific actions with requirements of the Resource Conservation and Recovery Act (RCRA) and the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), as well as routine monitoring to evaluate the quality of the water in the aquifer. The Snake River Plain aquifer is known to have been contaminated with tritium; however, the concentration dropped 93 percent between 1961 and 1994, possibly because of the elimination of tritium disposal, radioactive decay, and dispersion throughout the aquifer. Other known contaminants include cesium-137, iodine-129, strontium-90, and nonradioactive compounds such as trichloroethylene, chromium, and sodium. Components of nonradioactive waste have entered the aquifer as a result of past waste disposal practices. Elimination of groundwater injection, except for stormwater management and heat exchange, illustrates a change in disposal practices that has reduced the amount of these constituents in the groundwater. Information on recent groundwater monitoring and chemical analysis is presented in the annual site environmental report (Evans et al. 1998).

In 1997, INEEL used about 4.9 billion liters (1.3 billion gallons) per year from the Snake River Plain aquifer, the only source of water at INEEL (DOE 1999a). This represents less than 0.3 percent of the groundwater withdrawn from that aquifer. DOE holds a Federal Reserved Water Right for the INEEL site that permits a pumping capacity of 2.3 cubic meters (80 cubic feet) per second with a maximum water consumption of 43 billion liters (11.4 billion gallons) per year. INEEL's priority on water rights dates back to its establishment in 1950 (DOE 1996c).

All water used at ANL-W is groundwater from the Snake River Plain aquifer. The depth of the groundwater at ANL-W is approximately 195 meters (640 feet) and the flow is generally to the south-southwest. ANL-W uses approximately 188 million liters (49.6 million gallons) per year of water (ANL 1999b, Cascade Earth Sciences 1998).

No significant levels of radioactivity are found in the production wells at ANL-W. Constituents measured in the groundwater monitoring wells in 1997 were all below regulatory levels (ANL 1998b).

3.2.5 Geology and Soils

INEEL is located on the northwestern edge of the Eastern Snake River Plain that is bounded on the north and south by north to northwest-tending mountains and valley of the Basin and Range Provence (DOE 1999a). The upper 1 to 2 kilometers (0.6 to 1.2 miles) of the crust beneath INEEL is composed of interlayered basalt and sediment. The sediments are composed of fine-grained silts that were deposited by wind; silts, sands, and gravels deposited by streams; and clays, silts, and sands deposited in lakes. Rhyolitic (granite-like) volcanic rocks of unknown thickness lie beneath the basalt sediment sequence. The rhyolitic volcanic rocks erupted between 6.5 and 4.3 million years ago (Barghusen and Feit 1995). Lava tubes, which could have similar adverse effects as karst, occur in the INEEL area (Abbott, Crockett, and Moor 1997).

Within INEEL, economically viable sand, gravel, and pumice resources have been identified. Several quarries have supplied these materials to various onsite construction projects. Geothermal resources are potentially available in parts of the Eastern Snake River Plain, but neither of two boreholes drilled near INTEC encountered rocks with significant geothermal potential.

The Arco Segment of the Lost River Fault terminates about 12 kilometers (7.5 miles) from the INEEL boundary. The South Creek Segment of the Lemhi Fault terminates at the northwest boundary of the site. Both segments are considered capable (Abbott, Crockett, and Moor 1997). A capable fault is one that has had movement at or near the ground surface at least once within the past 35,000 years, or recurrent movement within the past 500,000 years.

The only recent earthquake activity recorded on or in the immediate vicinity of INEEL has been confined to several small- magnitude microearthquakes (less than 1.5 on the Richter Scale) (Jackson et al. 1993). The largest historic earthquake near INEEL took place in 1983, 107 kilometers (66 miles) to the northwest, near Borah Peak in the Lost River Range. The earthquake had a moment magnitude of 6.9 with a ground acceleration of 0.022 g to 0.078 g at INEEL (Jackson 1985). An earthquake with a maximum horizontal acceleration of 0.15 g is calculated to have an annual probability of occurrence of 1 in 5,000 at a central INEEL location (Barghusen and Feit 1995).

Basaltic volcanic activity occurred near INEEL from about 4 million to 2,100 years ago. Although no eruptions have occurred on the Eastern Snake River Plain during recorded history, lava flows of the Hell's Half Acre lava field erupted near the southern INEEL boundary as recently as 5,400 years ago. The most recent eruptions within the site area occurred about 2,100 years ago 30 kilometers (19 miles) southwest of the site at the Craters of the Moon Wilderness Area. Five volcanic zones have been identified on INEEL. The estimated recurrence interval for volcanism in these zones ranges from 16,000 to 100,000 years (Hackett and Smith 1994).

Four basic soilscapes exist on INEEL: river-transported sediments deposited on alluvial plains; fine-grained sediments eroded into lake or playa basins; colluvial sediments originating from bordering mountains; and wind-blown sediments over lava flows. The alluvial deposits follow the courses of the modern Big Lost River and Birch Creek. The playa soils are located in the north-central part of the INEEL site. The colluvial sediments are located along the western edge of the site. Wind-blown sediments (silt and sand) covering lava plains occupy the rest of the site's landscape (LMITCO 1997). The thickness of surficial sediments ranges from less than 0.3 meters (1 foot) at basalt outcrops east of INTEC to 95 meters (313 feet) near the Big Lost River sinks (DOE 1999a). No prime farmland lies within the INEEL boundaries (DOE 1999h).

The nearest capable fault to ANL-W is the South Creek Segment of the Lemhi Fault, which is located 31 kilometers (19 miles) northwest of the site (Abbott, Crockett, and Moor 1997). ANL-W is located within the Axial Volcanic Zone, which has an estimated recurrence interval for volcanism of 16,000 years (Hackett and Smith 1994). The site is situated within a topographically closed basin. Low ridges of basalt found east of the area rise as high as 30 meters (100 feet) above the level of the plain. Sediments cover most of the underlying basalt on the plain, except where pressure ridges form basalt outcrops (ANL 1999a). Soils in the ANL-W area have been found to resemble the Pancheri-Polatis-Tenno series, which generally consists of light brown-gray well-drained silty loams to brown extremely stony loams (ANL 1998a, DOA 1973). Soils are highly disturbed within developed areas of the site.

3.2.6 Ecological Resources

Ecological resources include terrestrial resources, wetlands, aquatic resources, and threatened and endangered species. Material presented in this section, unless otherwise noted, is from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c).

3.2.6.1 Terrestrial Resources

INEEL lies in a cool desert ecosystem dominated by shrub-steppe communities. Most land within the site is relatively undisturbed and provides habitat for species native to the region. The importance of this habitat was recently recognized when approximately 29,950 hectares (74,000 acres) located in the north central part of the site were designated as the INEEL Sagebrush Steppe Ecosystem Reserve (Figure 3–1) (DOE 1999g). Facilities and operating areas occupy only 2 percent of INEEL. Although sagebrush communities occupy about 80 percent of INEEL, a total of 20 plant communities have been identified (**Figure 3–4**). The interspersion

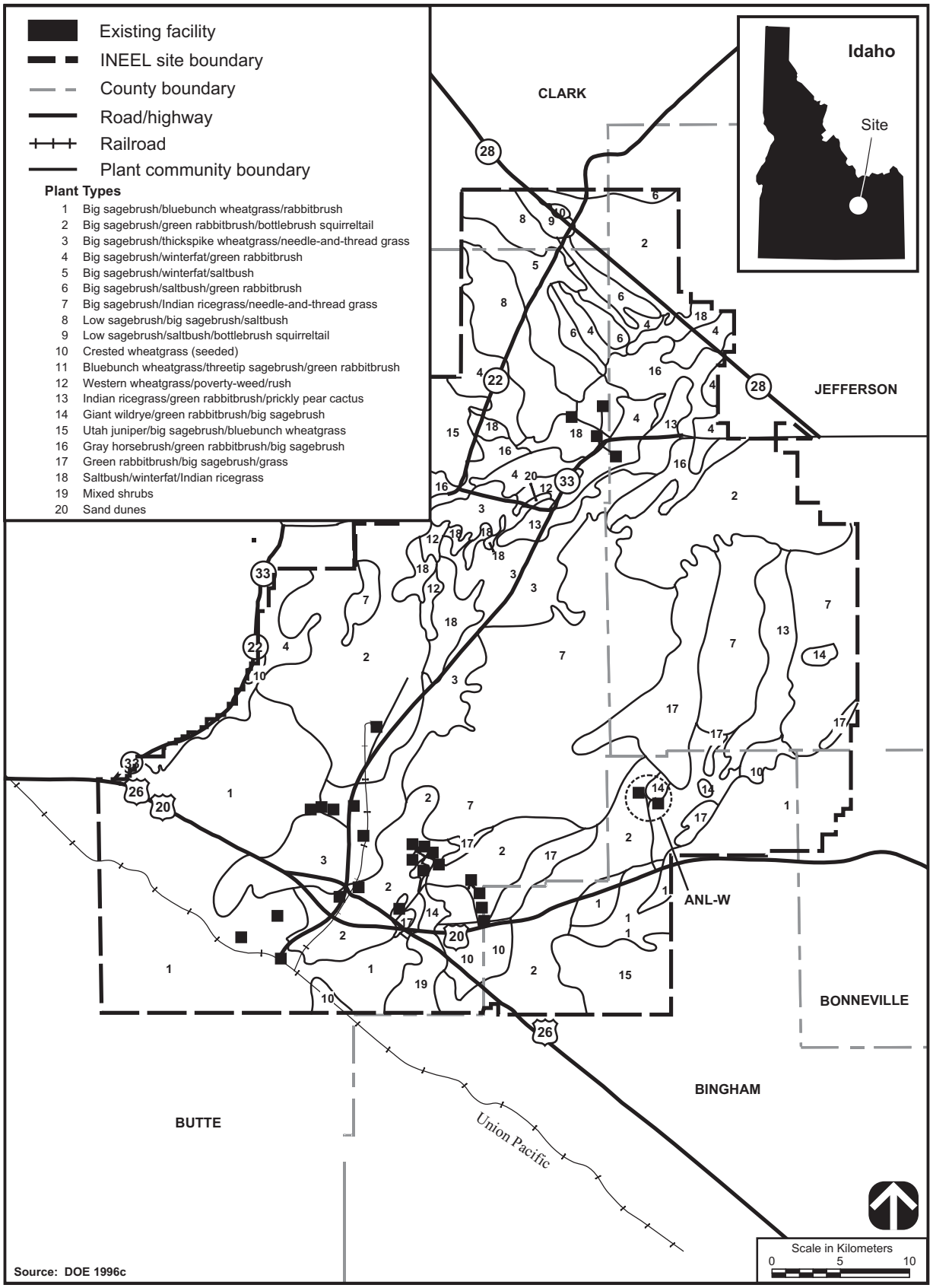

Figure 3–4 Distribution of Plant Communities at INEEL

of low and big sagebrush communities in the northern portion of INEEL and juniper communities located in the northwestern and southeastern portions of the site are considered sensitive habitats. The former provides critical winter and spring range for sage grouse and pronghorn, while the latter is important to nesting raptors and songbirds. Riparian vegetation, primarily cottonwood and willow, along the Big Lost River and Birch Creek also provides nesting habitat for hawks, owls, and songbirds. In total, 398 plant taxa have been documented on INEEL.

INEEL supports numerous animal species, including 2 amphibian, 11 reptile, 225 bird, and 44 mammal species (ESRF 1999). Common animals on the INEEL site include the short-horned lizard, gopher snake, sage sparrow, Townsend's ground squirrel, and black-tailed jackrabbit. Important game animals include the sage grouse, mule deer, elk, and pronghorn. During some winters, 4,500 to 6,000 pronghorn, or about 30 percent of Idaho's total population, may be found on the INEEL site. Pronghorn wintering areas are located in the northeastern portion of the site, in the area of the Big Lost River sinks, in the west-central portion of the site along the Big Lost River, and in the south-central portion of the site (DOE 1996c). Hunting of pronghorn and elk to control crop damage is permitted on site within 0.8 kilometers (0.5 miles) of the site boundary (LMITCO 1997). Numerous raptors, such as the golden eagle and prairie falcon, and carnivores, such as the coyote and mountain lion, also are found on the INEEL site.

ANL-W is located within one of several sagebrush communities found on the INEEL site (Figure 3–4). While sagebrush is present on undeveloped portions of the site, developed areas are nearly devoid of vegetation. Wildlife use of developed portions of the site is negligible; however, surrounding areas do provide natural habitat for a variety of wildlife. While elk and mule deer are the most important large mammals present in the area, many of the common species discussed above also would be expected. The ANL-W wastewater pond acts as an important source of water for wildlife found in the vicinity of the site (Cieminski and Flake 1995).

3.2.6.2 Wetlands

National Wetland Inventory maps prepared by the U.S. Fish and Wildlife Service indicate that the primary wetland areas on the INEEL site are associated with the Big Lost River, the Big Lost River spreading areas, and the Big Lost River sinks (or playas) (Figure 3–2). Smaller isolated wetlands (less than 0.4 hectares [1 acre]) also occur on the site (DOE 1996c). The only area of jurisdictional wetland is the Big Lost River sinks (Evans et al. 1998).

Wetland vegetation exists along the Big Lost River, which is located 18 kilometers (11 miles) west of ANL-W; however, this vegetation is in poor condition because of recent years of only intermittent flows. The Big Lost River spreading areas and Big Lost River sinks are seasonal wetlands and are located 34 kilometers (21 miles) west-southwest and 23 kilometers (14 miles) northwest of ANL-W, respectively. These areas can provide more than 809 hectares (2,000 acres) of wetland habitat during wet years. Within ANL-W itself, small areas of intermittent marsh occur along cooling tower blowdown ditches (Morris 1996).

3.2.6.3 Aquatic Resources

Aquatic habitat on the INEEL site is limited to the Big Lost River, Little Lost River, Birch Creek, and a number of liquid-waste disposal ponds. All three streams are intermittent and drain into four sinks in the north-central part of the site. Six species of fish have been observed within water bodies located on the site (ESRF 1999). Species observed in the Big Lost River include brook trout, rainbow trout, mountain whitefish, speckled dace, shorthead sculpin, and kokanee salmon. The Little Lost River and Birch Creek enter INEEL only during periods of high flow. Surveys of fish in these surface water bodies have not been conducted. The liquid waste disposal ponds on the INEEL site, while considered aquatic habitat, do not support fish.

There is no natural aquatic habitat on or in the vicinity of the ANL-W site. The nearest such habitat is the Big Lost River, which is located 18 kilometers (11 miles) west of the site. ANL-W waste disposal ponds do not contain any fish populations, but do provide habitat for a variety of aquatic invertebrates (Cieminski and Flake 1995).

3.2.6.4 Threatened and Endangered Species

Nineteen federally and state-listed threatened, endangered, and other special status species may be found on and in the vicinity of the INEEL site, 12 of which have been observed at the site (see Table 3–1 of the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* [DOE 1996c]). Two of these species are federally and/or state-listed. The bald eagle is listed by the U.S. Fish and Wildlife Service as threatened and by the State of Idaho as endangered. The peregrine falcon is listed by the state as endangered. The bald eagle rarely has been seen in the western and northern portions of INEEL. The peregrine falcon is an infrequent visitor to the site. The occurrence of the gray wolf (listed endangered, experimental populations) on the INEEL site is unverified. No critical habitat for threatened or endangered species, as defined in the Endangered Species Act, exists on the INEEL site.

The ANL-W area was surveyed in 1996 for threatened, endangered, and special status species (Morris 1996). The only listed species observed were the peregrine falcon and the loggerhead shrike. While no peregrine falcon nests were found near ANL-W, one perigrene falcon was observed perched on a power line 1.5 kilometers (0.9 miles) from the site. The loggerhead shrike, which is listed by Idaho as a species of concern, has been seen on numerous occasions in the vicinity of the site. The gray wolf (state endangered) and the pigmy rabbit and Townsend's big-eared bat (state species of concern) were not identified in the vicinity of ANL-W during the surveys. In addition, no federally or state-listed plants were found in the vicinity of the site. Consultation has been conducted with both the U.S. Fish and Wildlife Service and the state.

3.2.7 Cultural and Paleontological Resources

Cultural resources are human imprints on the landscape and are defined and protected by a series of Federal laws, regulations, and guidelines. INEEL has a well-documented recording of cultural and paleontological resources. Guidance for the identification, evaluation, and management of these resources is included in the *Idaho National Engineering Laboratory Management Plan for Cultural Resources (Final Draft)* (Miller 1995). Past studies, which covered 4 percent of the site, identified 1,506 cultural resource sites and isolated finds, including 688 prehistoric sites, 38 historic sites, 753 prehistoric isolates, and 27 historic isolates (DOE 1996c). As of January 1998, approximately 7 percent of INEEL has been surveyed, raising the number of potentially significant archaeological sites to 1,839 (DOE 1999a). Most surveys have been conducted near major facility areas in conjunction with modification, demolition, or abandonment of site facilities.

3.2.7.1 Prehistoric Resources

Prehistoric resources are physical properties remaining from human activities that predate written records. Prehistoric resources identified at INEEL are generally reflective of Native American hunting and gathering activities. Resources appear to be concentrated along the Big Lost River and Birch Creek, atop buttes, and within craters or caves. They include residential bases, campsites, caves, hunting blinds, rock alignments, and limited-activity locations such as lithic and ceramic scatters, hearths, and concentrations of fire-affected rock. Most sites have not been evaluated formally for nomination to the National Register, but are considered to be potentially eligible. Given the rather high density of prehistoric sites at INEEL, additional sites are likely to be identified as surveys continue.

The most recent cultural resource survey conducted near ANL-W took place in 1996 and covered an area to the south of the site that had been burned over by a wildfire and was proposed for revegetation (CEEA 1996).

A total of 12 isolated finds and 2 archaeological sites were located. Isolated finds include items such as pieces of Shoshone brownware pottery and projectile points. The archaeological sites include projectile points, scrappers, and volcanic glass flakes. Areas within the fenced portion of ANL-W are highly disturbed and are not likely to yield significant archaeological material.

3.2.7.2 Historic Resources

Thirty-eight historic sites and 27 historic isolates have been identified at INEEL. These resources are representative of European-American activities, including fur trapping and trading, immigration, transportation, mining, agriculture, and homesteading, as well as more recent military and scientific/engineering research and development activities. Examples of historic resources include Goodale's Cutoff (a spur of the Oregon Trail), remnants of homesteads and ranches, irrigation canals, and a variety of structures from the World War II era. EBR-I, the first reactor to achieve a self-sustaining chain reaction using plutonium instead of uranium as the principal fuel component, is listed on the National Register and is designated a National Historic Landmark. Many other INEEL structures built between 1949 and 1974 are considered eligible for the National Register because of their exceptional scientific and engineering significance and their major role in the development of nuclear science and engineering since World War II. Additional historic sites are likely to exist in unsurveyed portions of INEEL.

A number of recent items including farm implements, a belt buckle, broken glass, and a large scattering of cans have been found in the vicinity of ANL-W (CEEA 1996). EBR-II has been designated as an American Nuclear Society Historical Landmark (DOE 1997c). Consultation has been conducted with the State Historic Preservation Office.

3.2.7.3 Native American Resources

Native American resources at INEEL are associated with the two groups of nomadic hunters and gatherers that used the region at the time of European-American contact: the Shoshone and Bannock. Both of these groups used the area that now encompasses INEEL as they harvested plant and animal resources and obsidian from Big Southern Butte or Howe Point. Because INEEL is considered part of the Shoshone-Bannock Tribes' ancestral homeland, it contains many localities that are important for traditional, cultural, educational, and religious reasons. This includes not only prehistoric archaeological sites, which are important in a religious or cultural heritage context, but also features of the natural landscape and air, plant, water, or animal resources that have special significance. The value of certain areas on the INEEL site was recognized in the *1994 Memorandum of Agreement with the Shoshone-Bannock Tribes* (DOE 1994a), which provides Tribal members access to the Middle Butte area to perform sacred or religious ceremonies or other educational or cultural activities.

Although prehistoric Native American resources have been found in the vicinity of ANL-W (see Prehistoric Resources), the 1994 Memorandum of Agreement with the Shoshone-Bannock Tribes (DOE 1994a) does not affect the site (DOE 1997c). Consultation has been conducted with the Shoshone and Bannock Tribes.

3.2.7.4 Paleontological Resources

Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geologic age. The region encompassing INEEL has abundant and varied paleontological resources, including plant, vertebrate, and invertebrate remains from soils; lake and river sediments; and organic materials found in caves and archaeological sites. Vertebrate fossils recovered from the Big Lost River flood plain consist of isolated bones or teeth from large mammals of the Pleistocene or Ice Age. Fossils have been recorded in the vicinity of the Naval Reactors Facility, and a single mammoth tooth was salvaged during the excavation of a percolation pond immediately south of INTEC. Occasional fossil mammoth, horse, and camel skeletal

elements have been retrieved from the Big Lost River diversion dam and the Radioactive Waste Management Complex on the southwestern side of INEEL, and from river and alluvial fan gravels and Lake Terreton sediments near Test Area North (Abbott, Crockett, and Moor 1997). In total, 24 paleontological localities have been identified at INEEL (Miller 1995).

Paleontological resources were not found in the immediate vicinity of ANL-W during a recent archaeological survey (CEEA 1996).

3.2.8 Socioeconomics

Statistics for employment and economy are presented for the regional economic area, which encompasses 13 counties around INEEL located in Idaho and Wyoming. Statistics for population and housing, community services, and local transportation are presented for the region of influence. The region of influence is a 4-county area in Idaho in which 94.4 percent of all INEEL employees reside (**Table 3–7**). In 1997, the total INEEL employment was 8,291 persons (5.5 percent of the regional economic area civilian labor force).

Table 3–7 Distribution of Employees by Place of Residence in the INEEL Region of Influence, 1997

County	*Number of Employees*	*Total Site Employment (Percent)*
Bonneville	5,553	67
Bingham	1,077	13
Bannock	615	7.4
Jefferson	583	7
Region of influence total	**7,828**	**94.4**

Source: DOE 1999h.

3.2.8.1 Regional Economy Characteristics

Between 1990 and 1996, the civilian labor force in the regional economic area increased 26 percent to the 1996 level of 150,403. In 1996, the annual unemployment average in the regional economic area was 4.8 percent, slightly less than the annual unemployment average for Idaho (5.2 percent) and Wyoming (5 percent).

In 1995, service activities represented the largest sector of employment in the regional economic area (27.1 percent). This was followed by retail trade (20.4 percent) and government (19.5 percent). The totals for these employment sectors in Idaho were 21.5 percent, 19.6 percent, and 18.7 percent, respectively. The totals for these employment sectors in Wyoming were 21.1 percent, 20.8 percent, and 25 percent, respectively.

3.2.8.2 Population and Housing

In 1996, the region of influence population totaled 213,547. Between 1990 and 1996 the region of influence population increased by 10.6 percent, compared with a 17.5 percent increase in Idaho's population. Between 1980 and 1990 the number of housing units in the region of influence increased by 6.7 percent, compared with a 10.2 percent increase in Idaho (DOE 1999h). The total number of housing units in the region of influence for 1990 was 69,760. In 1995, the total number of owner and renter housing units within the region of influence was 74,600 (DOE 1996a). The 1990 region of influence homeowner vacancy rate was 2.1 percent, compared with Idaho's rate of 2 percent. The region of influence renter vacancy rate was 8.3 percent, compared with Idaho's rate of 7.3 percent.

3.2.8.3 Community Services

Community services include public education and public safety. In 1997, school districts providing public education in the INEEL region of influence were operating at capacities of between 50 to 100 percent. Total student enrollment in the INEEL region of influence in 1997 was 50,168, and the student-to-teacher ratio averaged 18.8 to 1. In 1990, the average student-to-teacher ratio for Idaho was 12.8 to 1. In 1997, a total of 475 sworn police officers were serving the four-county region of influence. The average INEEL region of influence officer-to-population ratio was 2.2 officers per 1,000 persons. This compares with the 1990 state average of 1.5 officers per 1,000 persons.

3.2.8.4 Local Transportation

Vehicular access to INEEL is provided by U.S. Routes 20 and 26 to the south and State Routes 22 and 33 to the north. U.S. Routes 20 and 26 and State Routes 22 and 33 all share rights-of-way west of INEEL (Figure 3–1). DOE shuttle vans provide transportation between INEEL facilities and Idaho Falls for DOE and contractor personnel. The major railroad in the region of influence is the Union Pacific Railroad. The railroad's Blackfoot-to-Arco Branch provides rail service to the southern portion of INEEL. A DOE-owned spur connects the Union Pacific Railroad to INEEL by a junction at Scovill Siding. There are no navigable waterways within the region of influence capable of accommodating waterborne transportation of material shipments to INEEL. Fanning Field in Idaho Falls and Pocatello Municipal Airport in Pocatello provide jet air passenger and cargo service for both national and local carriers.

3.2.9 Environmental Justice

Under Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations,* Federal agencies are responsible for identifying and addressing the possibility of disproportionately high and adverse health or environmental effects of programs and policies on minority and low-income populations in potentially affected areas. Minority populations refer to all people of color, exclusive of white non-Hispanics. Low-income populations refer to households whose incomes are below the Federal poverty threshold. In the case of INEEL, the potentially affected area includes only parts of central Idaho.

The 1990 census data show that the percentage of minorities within the contiguous United States was 24.1 percent, while within the State of Idaho it was 7.7 percent. The data also show that 13.1 percent of the incomes within the United States were below the poverty threshold. Within Idaho, 13.3 percent of the incomes were below the poverty threshold.

The potentially affected area surrounding ANL-W is defined by a circle with an 80-kilometer (50-mile) radius centered at latitude 43°35'41.7" N, longitude 112°39'18.7" W. The total population residing within that area in 1990 was 180,582. The proportion of this population that was considered minority was 8.7 percent. At the time of the 1990 census, Hispanics and Native Americans were the largest minority groups within that area, constituting 5.2 percent and 2.2 percent of the total population, respectively. Asians constituted about 1 percent and Blacks about 0.3 percent.

A breakdown of incomes in the potentially affected area also is available from the 1990 census data. At that time, the poverty threshold was $9,981 for a family of three with one related child under 18 years of age. A total of 25,046 persons (15 percent of the total population) residing within the potentially affected area around ANL-W reported incomes below that threshold.

3.2.10 Existing Human Health Risk

Public and occupational health and safety issues include the determination of potentially adverse effects on human health that result from acute and chronic exposures to ionizing radiation and hazardous chemicals.

3.2.10.1 Radiation Exposure and Risk

Major sources and levels of background radiation exposure to individuals in the vicinity of INEEL are shown in **Table 3–8**. Annual background radiation doses to individuals are expected to remain constant over time. The total dose to the population, in terms of person-rem, changes as the population size changes. Background radiation doses are unrelated to INEEL operations.

Table 3–8 Sources of Radiation Exposure to Individuals in the INEEL Vicinity Unrelated to INEEL Operations

Source	*Effective Dose Equivalent (millirem per year)*
Natural Background Radiation [a]	
Cosmic radiation	48
External terrestrial radiation	74
Internal terrestrial/cosmogenic radiation	40
Radon in homes (inhaled)	200 [b, c]
Other Background Radiation [c]	
Diagnostic x-rays and nuclear medicine	53
Weapons test fallout	less than 1
Air travel	1
Consumer and industrial products	10
Total	**427**

[a] Evans et al. 1998.
[b] An average for the United States.
[c] NCRP 1987.

Releases of radionuclides to the environment from INEEL operations provide another source of radiation exposure to individuals in the vicinity of INEEL. Types and quantities of radionuclides released from INEEL operations in 1997 are listed in the *Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1997* (Evans et al. 1998). The doses to the public resulting from these releases are presented in **Table 3–9.** These doses fall within radiological limits per DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, and are much lower than those of background radiation.

Using a risk estimator of 500 latent cancer deaths per 1 million person-rem to the public (see Appendix E), the fatal cancer risk to the maximally exposed member of the public due to radiological releases from INEEL operations in 1997 is estimated to be 1.1×10^{-8}. That is, the estimated probability of this person dying of cancer at some point in the future from radiation exposure associated with one year of INEEL operations is less than 2 in 100 million. (It takes several to many years from the time of radiation exposure for a cancer to manifest itself.)

Table 3–9 Radiation Doses to the Public From Normal INEEL Operations in 1997 (Total Effective Dose Equivalent)

Members of the Public	*Atmospheric Releases*		*Liquid Releases*		*Total*	
	Standard [a]	*Actual*	*Standard* [a]	*Actual*	*Standard* [a]	*Actual*
Maximally exposed offsite individual (millirem)	10	0.021	4	0	100	0.021
Population within 80 kilometers (50 miles) (person-rem) [b]	None	0.23	None	0	100	0.23
Average individual within 80 kilometers (50 miles) (millirem) [c]	None	0.0019	None	0	None	0.0019

[a] The standards for individuals are given in DOE Order 5400.5. As discussed in that Order, the 10-millirem per year limit from airborne emissions is required by the Clean Air Act, and the 4-millirem per year limit is required by the Safe Drinking Water Act. For this EIS, the 4-millirem per year value is assumed conservatively to be the limit for the sum of doses from all liquid pathways. The total dose of 100 millirem per year is the limit from all pathways combined. The 100-person-rem value for the population is given in the proposed 10 CFR 834, *Radiation Protection of the Public and Environment; Proposed Rule*, as published in 58 FR 16268. If the potential total dose exceeds the 100 person-rem value, the contractor operating the facility is required to notify DOE.

[b] About 121,400 in 1997.

[c] Obtained by dividing the population dose by the number of people living within 80 kilometers (50 miles) of the site.

Source: Evans et al. 1998.

According to the same risk estimator, 0.0012 excess latent fatal cancers are projected in the population living within 80 kilometers (50 miles) of INEEL from normal operations in 1997. To place this number in perspective, it may be compared with the number of fatal cancers expected in the same population from all causes. The 1995 mortality rate associated with cancer for the entire U.S. population was 0.2 percent per year. Based on this mortality rate, the number of latent fatal cancers expected during 1997 from all causes in the population living within 80 kilometers (50 miles) of INEEL was 243. This expected number of fatal cancers is much higher than the 0.00012 fatal cancers estimated from INEEL operations in 1997.

INEEL workers receive the same dose as the general public from background radiation, but they also receive an additional dose from working in facilities with nuclear materials. The average dose to the individual worker and the cumulative dose to all workers at INEEL from operations in 1997 are presented in **Table 3–10.** These doses fall within the radiological regulatory limits of 10 CFR 835 (DOE 1995a). According to a risk estimator of 400 latent fatal cancers per 1 million person-rem among workers (see Appendix E), the number of projected fatal cancers among INEEL workers from normal operations in 1997 is 0.046. The risk estimator for workers is lower than the estimator for the public because of the absence from the work force of the more radiosensitive infant and child age groups.

Table 3–10 Radiation Doses to Workers From Normal INEEL Operations in 1997 (Total Effective Dose Equivalent)

Occupational Personnel	*Onsite Releases and Direct Radiation*	
	Standard [a]	*Actual*
Average radiation worker (millirem)	None [b]	101 [c]
Total workers (person-rem) [d]	None	115 [c]

[a] The radiological limit for an individual worker is 5,000 millirem per year. However, DOE's goal is to maintain radiological exposure as low as is reasonably achievable. Therefore, DOE has established an administrative control level of 2,000 millirem per year (DOE Order N 441.1); the site must make reasonable attempts to maintain individual worker doses below this level.

[b] No standard is specified for an "average radiation worker"; however, the maximum dose that this worker may receive is limited to that given in footnote "a."

[c] Does not include doses received at the Naval Reactors Facility. The impacts associated with this facility fall under the jurisdiction of the Navy as part of the Nuclear Propulsion Program.

[d] 1,141 workers with measurable doses in 1997.

Sources: DOE 1995a, DOE 1998d.

A more detailed presentation of the radiation environment, including background exposures and radiological releases and doses, is presented in the *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 1997* (Evans et al. 1998). The concentrations of radioactivity in various environmental media (including air, water, and soil) in the site region (on and off the site) also are presented in that report.

External radiation doses and concentrations of plutonium in air have been measured at ANL-W. The onsite dose is measured for comparison against natural background levels measured at offsite control locations; the numerical differences in these measurements may be directly attributable to radiological sources that are located in the vicinity of the onsite measurement location. In 1997, the annual average dose within the area was about 144 millirem. This is about 5 millirem higher than the average dose measured at offsite control locations. Concentrations in air of plutonium-239 and plutonium-240 in 1996 were 3.4×10^{-18} microcuries per milliliter. This value is essentially the same as those measured at an offsite control location. Finally, concentrations in air of gross alpha and beta radiation at ANL-W are 6.0×10^{-16} microcuries per milliliter and 2.0×10^{-14} microcuries per milliliter, respectively. These alpha and beta radiation concentrations are essentially the same as those measured at offsite control locations (Evans et al. 1998).

3.2.10.2 Chemical Environment

Table 3–2 identifies the hazardous (i.e., carcinogenic and toxic/noncarcinogenic) chemicals that are emitted to the air at INEEL. The list includes only those chemicals that have ambient air quality standards and would be emitted under any one of the alternatives analyzed in this EIS. These include 1,3-butadiene, acetaldehyde, acrolein, benzene, cadmium, formaldehyde, toluene, and xylene.

Health impacts on the public may occur by inhaling air containing hazardous chemicals, ingesting contaminated drinking water or food, and direct exposure (skin contact). The primary health impacts from exposure to hazardous chemicals are from inhalation. Two major health effects are observed from the listed chemicals, the carcinogenic effect and the noncarcinogenic effect. These are presented below.

Carcinogenic Effects: These effects are estimated as the incremental probability of an individual developing cancer over a lifetime as a result of exposure to the potential carcinogen. This could be incremental or excess individual lifetime cancer risk.

Noncarcinogenic Effects: These effects are determined by the ratio between the calculated, or measured, concentration of the chemical in the air and the reference concentration or dose. This ratio is known as the hazard quotient. Hazard quotients for noncarcinogens are summed to obtain the hazard index. If the hazard index is less than 1, then no adverse health effects are expected.

For some chemicals where the weight of evidence is weak and carcinogenecity is not well established, the impacts of both cancer and noncancer effects are determined. **Table 3–11** summarizes the baseline hazardous chemical impacts to the public. This table lists only those chemicals for which reference concentrations for cancer or toxicity are available from the Integrated Risk Information System. The baseline concentrations are estimates of the highest existing concentrations and represent the highest concentrations to which individuals from the public could be exposed under normal operations (excluding accident conditions). These concentrations are in compliance with applicable guidelines. Additional information on estimating the health impacts of hazardous chemicals is presented in Appendix E, Section E.5.

The exposure of workers to hazardous chemicals varies among facilities and the operational activities, and the available information is insufficient for a meaningful estimate of impacts. Workers are protected by adherence to the Occupational Health and Safety Administration (OSHA) and EPA standards that regulate workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. Monitoring the frequency

and the amount of chemicals released in operational processes ensures that these standards are not exceeded. Further, DOE requires that the environment in the workplace be as free as possible from recognized hazards that cause, or are likely to cause, illness or physical harm. Therefore, workplace conditions at INEEL are substantially better than required by standards.

Table 3–11 Hazardous Chemical Impacts to the Public from Existing Activities at INEEL

Chemical	*Hazard Quotient*	*Cancer Risk*
1,3-Butadiene	Not applicable	2.8×10^{-7}
Acetaldehyde	0.0002	2.4×10^{-8}
Acrolein	0.01340	Not applicable
Cadmium	Not applicable	7.5×10^{-8}
Benzene	Not applicable	2.4×10^{-7}
Formaldehyde	Not applicable	1.6×10^{-7}
Toluene	Less than 0.01	Not applicable
Hazard Index	Less than 0.0236	Not applicable

Source: DOE 1995a, or dispersion modeling.

3.2.10.3 Health Effects Studies

Epidemiological studies were conducted on communities surrounding INEEL to determine whether there are excess cancers in the general population. Two of these are described in more detail in Appendix M.4.4 of the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c). No excess cancer mortality was reported, and although excess cancer incidence was observed, no association with INEEL was established. A study by the State of Idaho completed in June 1996 found excess brain cancer incidence in the six counties surrounding INEEL, but a follow-up survey concluded that, "There was nothing that clearly linked all these cases to one another or any one thing" (DOE 1996c).

No occupational epidemiological studies have been completed at INEEL to date, but several worker health studies were initiated recently at INEEL and another is almost complete. Researchers from the Boston University School of Public Health, in cooperation with the National Institute of Occupational Safety and Health, are investigating the effects of work force restructuring (downsizing) in the nuclear weapons industry. The health of displaced workers will be studied. Under a National Institute of Occupational Safety and Health cooperative agreement, the epidemiologic evaluation of childhood leukemia and paternal exposure to ionizing radiation now includes INEEL as well as other DOE sites. Another study begun in October 1997, *Medical Surveillance for Former Workers at INEEL*, is being carried out by a group of investigators consisting of the Oil, Chemical, and Atomic Workers International Union; Mount Sinai School of Medicine; the University of Massachusetts at Lowell; and Alice Hamilton College. A mortality study of the work force at INEEL being conducted by National Institute of Occupational Safety and Health is pending publication. DOE has implemented an epidemiologic surveillance program to monitor the health of current INEEL workers. A discussion of this program is given in Appendix M.4.4 of the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c).

3.2.10.4 Accident History

DOE conducted a study, the *Idaho National Engineering Laboratory Historical Dose Evaluation,* to estimate the potential offsite radiation doses for the entire operating history of INEEL (DOE 1996c). Releases resulted from a variety of tests and experiments as well as a few accidents at INEEL. The study concluded that these releases contributed to the total radiation dose during test programs of the 1950s and early 1960s. The frequency and size of releases has declined since that time. There have been no serious unplanned or

accidental releases of radioactivity or other hazardous substances at INEEL facilities in the last 10 years of operation.

3.2.10.5 Emergency Preparedness

Each DOE site has established an emergency management program that would be activated in the event of an accident. This program has been developed and maintained to ensure adequate response to most accident conditions and to provide response efforts for accidents not specifically considered. The emergency management program includes emergency planning, training, preparedness, and response.

Government agencies whose plans are interrelated with the INEEL Emergency Plan for Action include the State of Idaho; Bingham, Bonneville, Butte, Clark, and Jefferson counties; the Bureau of Indian Affairs; and the Fort Hall Indian Reservation. INEEL contractors are responsible for responding to emergencies at their facilities. Specifically, the emergency action director is responsible for recognition, classification, notification, and protective action recommendations. At INEEL, emergency preparedness resources include fire protection from onsite and offsite locations and radiological and hazardous chemical material response. Emergency response facilities include an emergency control center at each facility, at the INEEL warning communication center, and at the INEEL site emergency operations center. Seven INEEL medical facilities are available to provide routine and emergency service. In addition, DOE has specified actions to be taken at all DOE sites to implement lessons learned from the emergency response to an accidental explosion at Hanford in May 1997.

3.2.11 Waste Management

Waste management includes minimization, characterization, treatment, storage, transportation, and disposal of waste generated from ongoing DOE activities. The waste is managed using appropriate treatment, storage, and disposal technologies, and is in compliance with all applicable Federal and state statutes and DOE Orders.

3.2.11.1 Waste Inventories and Activities

INEEL manages the following types of waste: high-level radioactive, transuranic, mixed transuranic, low-level radioactive, mixed, hazardous, and nonhazardous. Waste generation rates and the inventory of stored waste from activities at INEEL are provided in **Table 3–12.** The INEEL waste management capabilities are summarized in **Table 3–13.** More detailed descriptions of the waste management system capabilities at INEEL are included in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c) and the Programmatic Spent Nuclear Fuel EIS (DOE 1995a).

The EPA placed INEEL on the National Priorities List[3] on December 21, 1989. In accordance with CERCLA, DOE entered into a Consent Order with the EPA and the State of Idaho to coordinate cleanup activities at INEEL under one comprehensive strategy. This agreement integrates DOE's CERCLA response obligations with RCRA corrective action obligations. Aggressive plans are in place to achieve early remediation of sites that represent the greatest risk to workers and the public. The goal is to complete remediation of contaminated sites at INEEL to support delisting from the National Priorities List by the year 2019 (DOE 1996c). More information on regulatory requirements for waste disposal is provided in Chapter 5.

[3]*The National Priorities List is a list of those sites requiring cleanup that appear to have the most serious threat to public health or the environment due to the release of hazardous substances. The list is promulgated by the EPA under CERCLA.*

Table 3–12 Waste Generation Rates and Inventories at INEEL

Waste Type	*Generation Rate (cubic meters per year)*	*Inventory (cubic meters)*
High-level radioactive	0 [a]	4,000 [b]
Transuranic	0 [a, c]	65,000 [d]
Low-level radioactive	6,400 [e]	6,000 [f]
Mixed [g]	230	1,700
Hazardous	835 [c, h]	Not applicable [I]
Nonhazardous		
Liquid	2,000,000 [c, j]	Not applicable [I]
Solid	62,000 [c]	Not applicable [I]

[a] Refer to the text.
[b] INEEL 1999b. The inventory is calcined high-level radioactive waste.
[c] Moor and Peterson 1999.
[d] DOE 1995a.
[e] LMITCO 1998.
[f] Bright 1999.
[g] DOE 1998c.
[h] Includes 760 cubic meters that are recyclable.
[I] Generally, hazardous and nonhazardous waste is not held in long-term storage.
[j] Projected annual average generation amounts for 1997 to 2006.
Note: To convert from cubic meters to cubic yards, multiply by 1.31.
Sources: Given in footnotes b through g, above.

Table 3–13 Waste Management Capabilities at INEEL

Facility Name/Description	*Capacity*	*Status*	*Applicable Waste Type*						
			HLW	*TRU*	*Mixed TRU*	*LLW*	*Mixed*	*Haz*	*Non-Haz*
Treatment Facility (cubic meters per year except as otherwise specified)									
INTEC High-Efficiency Particulate Air Filter Leach, cubic meters per day	0.21	On-line			X		X		
INTEC Debris Treatment and Containment, cubic meters per day	88	Waiting for Part B Permit			X		X		
Advanced Mixed Waste Treatment Project	6,500	Planned for 2003			X		X		
INTEC New Waste Calcining Facility	248	On-line	X		X				
ANL–W Remote Treatment Facility	42	Planned for 2000		X	X	X	X		
ANL–W Hot Fuel Examination Facility Waste Characterization Area	37	On-line		X	X				
INTEC Waste Immobilization Facility	48	Planned for 2020			X	X	X		
INTEC Liquid Effluent Treatment and Disposal Facility	11,365	On-line					X		
INTEC High-Level Radioactive Waste Evaporator	6,138	On-line			X	X	X		
INTEC Process Equipment Waste Evaporator	13,000	On-line			X	X	X		

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed	Haz	Non-Haz
ANL–W Sodium Processing Facility	698	On-line					X		
Test Area North Cask Dismantlement	11	On-line					X		
Waste Reduction Operations Complex - Debris Sizing, kilograms per hour	1,149	Planned for 2000				X	X		
Waste Reduction Operations Complex - Macroencapsulation, kilograms per hour	2,257	Planned for 1999					X		
Waste Reduction Operations Complex - Stabilization, cubic meters per day	7.6	On-line					X		
Waste Experimental Reduction Facility	49,610	On-line				X	X	X	
INTEC Sewage Treatment Plant	3,200,000	On-line							X
Storage Facility (cubic meters)									
INTEC Tank Farm	12,533	On-line	X [a]		X		X		
INTEC Calcine Bin Sets	6,950	On-line	X						
ANL–W Radioactive Sodium Storage	75	On-line			X		X		
ANL–W Sodium Components Maintenance Shop	200	On-line					X		
ANL–W Radioactive Scrap and Waste Storage	193	On-line		X	X	X	X		
ANL–W EBR-II Sodium Boiler Drain Tank	64	On-line					X		
ANL–W Hot Fuel Examination Facility Waste Characterization Area	37	On-line		X	X				
INTEC Fluorinel Dissolution Process High-Efficiency Particulate Air Filter Storage	25	On-line			X		X		
INTEC New Waste Calcining Facility High-Efficiency Particulate Air Filter Storage	56	On-line			X		X		
INTEC Chemical Processing Plant-1619 Storage	45	On-line					X	X	
INTEC Chemical Processing Plant-1617 Staging	8,523	On-line					X	X	
Radioactive Waste Management Complex Transuranic Storage Area [b]	64,900	On-line		X	X	X	X		
Radioactive Waste Management Complex Waste Storage [b]	112,400	On-line		X	X	X	X		
Radioactive Waste Management Complex Intermediate-Level Storage	100	On-line		X					
Waste Reduction Operations Complex Power Burst Facility Mixed Waste Storage	129	On-line					X	X	

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed	Haz	Non-Haz
Portable storage at Special Power Excursion Reactor Test IV	237	On-line					X	X	
Power Burst Facility Waste Experimental Reduction Facility Waste Storage Building	685	On-line					X	X	
Test Area North 647 Waste Storage	104	On-line					X	X	
Test Area North 628 Specific Manufacturing Complex Container Storage	125	On-line					X	X	
Disposal Facility (cubic meters per year)									
Radioactive Waste Management Complex Disposal Facility	37,700	On-line				X			
Central Facilities Area Landfill Complex	48,000	On-line							X
Percolation ponds	2,000,000	On-line							X

HAZ = hazardous waste, HLW = high-level radioactive waste, LLW = low-level radioactive waste, TRU = transuranic waste

[a] Sodium-bearing waste.

[b] For these facilities, the low-level radioactive waste and mixed waste are considered alpha-contaminated low-level radioactive waste and alpha-contaminated mixed waste (waste containing between 10 and 100 nanocuries of alpha activity per gram).

Sources: DOE 1999f, DOE 1999h.

3.2.11.2 High-Level Radioactive Waste

High-level radioactive waste at INEEL was generated in the process of extracting useful isotopes from spent nuclear fuel at INTEC. Most of this fuel was from the Naval Reactors Program. Most aqueous solutions from spent nuclear fuel processing and isotope extraction were concentrated by evaporation and separated into low- and high-level radioactive waste streams in the Process Equipment Waste Evaporator. The liquid high-level radioactive waste was stored in subsurface tanks and then transformed by calcination into solid metal oxides in a granular form. This calcination was completed in February 1998. The calcine is stored in stainless steel bins in near-surface concrete vaults where it awaits further processing into a form suitable for emplacement in a Federal repository. INEEL will meet the requirements of a December 1991 Consent Order with the State of Idaho and the EPA to cease the use of existing storage tanks without constructing new tanks. Subsequently, the calcined waste will be treated to meet RCRA provisions on a schedule to be negotiated with the State of Idaho under the Federal Facilities Compliance Act.

Although sodium-bearing waste is not high-level radioactive waste as specified in the Nuclear Waste Policy Act of 1982, it has been managed historically as high-level radioactive waste at INEEL. This is because some of the physical and chemical properties of these two waste types are similar, e.g., both are acidic and both contain similar radionuclides, including transuranics (DOE 1999i). About 5,300 cubic meters (1.4×10^6 gallons) of liquid sodium-bearing waste remain in the INTEC Tank Farm. It is anticipated that this waste will be calcined in the New Waste Calcining Facility and then be treated to meet RCRA provisions.

3.2.11.3 Transuranic Waste

Transuranic waste generated since 1972 is segregated into contact-handled and remote-handled categories and stored at the Radioactive Waste Management Complex in a form designed for eventual retrieval (DOE 1996c).

Some transuranic waste also is stored at the Radioactive Scrap and Waste Facility at ANL–W (DOE 1995a). There is very little transuranic waste generated at INEEL. Most of the transuranic waste in storage was received from the Rocky Flats Environmental Technology Site (DOE 1996a). Transuranic waste currently is being stored pending shipment to the Waste Isolation Pilot Plant. The first shipment of transuranic waste from INEEL was received at the Waste Isolation Pilot Plant on April 28, 1999 (DOE 1999d). Transuranic waste is treated to meet the Waste Isolation Pilot Plant Waste Acceptance Criteria, packaged in accordance with DOE and U.S. Department of Transportation requirements, and transported to the Waste Isolation Pilot Plant for disposal (DOE 1996c).

The existing treatment facilities for transuranic waste at INEEL are limited to testing, characterization, and repackaging. The planned Waste Characterization Facility will characterize (identify) transuranic waste and either reclassify it (if it is found to be low-level radioactive waste) for disposal on the site, or prepare it so that it meets Waste Isolation Pilot Plant Waste Acceptance Criteria (DOE 1996c).

The Advanced Mixed Waste Treatment Project will be operated as a private sector treatment facility after its construction is completed. This facility will: (1) treat waste to meet Waste Isolation Pilot Plant Waste Acceptance Criteria, RCRA Land Disposal Restrictions, and required Toxic Substances Control Act standards; (2) reduce waste volume and life cycle cost to DOE; and (3) perform tasks in a safe and environmentally compliant manner (INEEL 1999a).

Waste containing between 10 and 100 nanocuries of alpha activity per gram of transuranic radionuclides is called alpha low-level radioactive waste. Although this waste technically is considered low-level radioactive waste rather than transuranic waste, it cannot be disposed of at INEEL because it does not meet all INEEL low-level radioactive waste disposal facility acceptance criteria. Alpha low-level radioactive waste and alpha mixed waste are managed together as part of the Transuranic Waste program. It is expected that this waste will be treated by the Advanced Mixed Waste Treatment Project and then disposed of at the Waste Isolation Pilot Plant (DOE 1999h).

3.2.11.4 Low-Level Radioactive Waste

Liquid low-level radioactive waste either is evaporated and processed to a calcine form or solidified before disposal (DOE 1996a). INTEC has the capability to treat aqueous low-level radioactive waste. Liquid low-level radioactive waste is concentrated at the INTEC Process Equipment Waste Evaporator, and the condensed vapor is processed by the Liquid Effluent Treatment and Disposal Facility. The concentrated materials remaining after evaporation are pumped to the INTEC Tank Farm. Some small volumes of liquid low-level radioactive waste are solidified at the Waste Experimental Reduction Facility for disposal at the Radioactive Waste Management Complex. In addition, small volumes of aqueous low-level radioactive waste are discharged to the double-lined pond at the Test Reactor Area for evaporation (DOE 1995a).

Most solid low-level radioactive waste at INEEL is sent to the Waste Experimental Reduction Facility for treatment by incineration, compaction, size reduction, or stabilization before shipment for disposal at the Radioactive Waste Management Complex or offsite disposal facilities (DOE 1999h). Disposal occurs in pits and concrete-lined soil vaults in the subsurface disposal area of the Radioactive Waste Management Complex (DOE 1995a). About 40 percent of the low-level radioactive waste generated at INEEL (containing less than 10 nanocuries per gram of radioactivity) is buried in shallow trenches; the remaining 60 percent is buried at the Radioactive Waste Management Complex following treatment for volume reduction. Additionally, some low-level radioactive waste is shipped off site to be incinerated, and the residual ash is returned to INEEL for disposal. The Radioactive Waste Management Complex is expected to be filled to capacity by the year 2030, although some proposals would close the Low-Level Radioactive Waste Disposal Facility by 2006 (DOE 1999h).

3.2.11.5 Mixed Waste

Mixed waste is divided into two categories for management purposes: alpha mixed waste and beta-gamma mixed waste. Most of the alpha mixed waste stored at INEEL is waste that has been reclassified from mixed transuranic waste and is managed as part of the transuranic waste program. Therefore, this section deals only with beta-gamma mixed waste (DOE 1995a).

Mixed waste, including polychlorinated biphenyls–contaminated low-level radioactive waste, is stored in several onsite areas awaiting the development of treatment methods (DOE 1996c). Mixed waste is stored at the mixed waste storage facility (the Waste Experimental Reduction Facility Waste Storage Building) and in portable storage units at the Power Burst Facility area. In addition, smaller quantities of mixed waste are stored in various facilities at INEEL, including the Hazardous Chemical/Radioactive Waste Facility at INTEC and the Radioactive Sodium Storage Facility and Radioactive Scrap and Waste Storage Facility at ANL-W (DOE 1995a). Although mixed waste is stored in many locations at INEEL, the bulk of that volume is solid waste stored at the Radioactive Waste Management Complex (DOE 1996c).

Aqueous mixed waste is concentrated at INTEC. The condensate from the waste evaporator is processed by the Liquid Effluent Treatment and Disposal Facility. The concentrated material remaining after evaporation (mixed waste) is pumped to the INTEC Tank Farm for storage (DOE 1999h).

As part of the site treatment plans required by the Federal Facilities Compliance Agreement, preferred treatment options have been identified to eliminate the hazardous waste component for many types of mixed waste (DOE 1995a). Mixed waste is being or will be processed to RCRA Land Disposal Restrictions treatment standards through several treatment facilities. Those treatment facilities and their operational status are: (1) Waste Experimental Reduction Facility Incinerator (operational); (2) Waste Experimental Reduction Facility Stabilization (operational); (3) Test Area North Cask Dismantlement (operational); (4) Sodium Process Facility (operational); (5) High-Efficiency Particulate Air Filter Leach (operational); (6) Waste Reduction Operations Complex Macroencapsulation (June 2000); (7) Waste Reduction Operations Complex Mercury Retort (June 2000); (8) Debris Treatment (September 2000); and (9) Advanced Mixed Waste Treatment Project (March 2003). Commercial treatment facilities also are being considered, as appropriate. Currently, limited amounts of mixed waste are disposed of at Envirocare of Utah (DOE 1999h).

3.2.11.6 Hazardous Waste

Approximately 1 percent of the total waste generated at INEEL is hazardous waste. Most of the hazardous waste generated annually at INEEL is transported off site for treatment and disposal (DOE 1995a). Offsite shipments are surveyed to determine that the waste has no radioactive content and therefore are not mixed waste (DOE 1996c). Highly reactive or unstable materials, such as waste explosives, are addressed on a case-by-case basis, and are either stored, burned, or detonated, as appropriate.

3.2.11.7 Nonhazardous Waste

Approximately 90 percent of the waste generated at INEEL is classified as industrial waste and is disposed of on site in a landfill complex in the Central Facilities Area and off site at the Bonneville County landfill (DOE 1995a). The onsite landfill complex contains separate areas for petroleum-contaminated media, industrial waste, and asbestos waste (DOE 1999h). The onsite landfill is 5 hectares (12 acres), and is being expanded by 91 hectares (225 acres) to provide capacity for at least 30 years (DOE 1996c).

Sewage is disposed of in surface impoundments in accordance with the terms of the October 7, 1992, Consent Order. Wastewater in the impoundments is allowed to evaporate, and the resulting sludge is placed in the landfill. Solids are separated and reclaimed where possible (DOE 1996c). Nonhazardous service wastewater

generated at INTEC is disposed of in percolation ponds at a flow rate of 3.8 million to 7.6 million liters (1 million to 2 million gallons) per day. The INTEC sanitary sewer system collects and transfers sanitary waste to the sewage treatment lagoons east of INTEC for treatment and disposal. This system has a capacity of 3,200,000 cubic meters (4,190,000 cubic yards) per year (DOE 1999h).

3.2.11.8 Waste Minimization

The DOE Idaho Operations Office has an active waste minimization and pollution prevention program to reduce the total amount of waste generated and disposed of at INEEL. This is accomplished by eliminating waste through source reduction or material substitution; by recycling potential waste materials that cannot be minimized or eliminated; and by treating all of the waste that is generated to reduce its volume, toxicity, or mobility prior to storage or disposal. The Idaho Operations Office published its first waste minimization plan in 1990, which defined specific goals, methodology, responsibility, and achievements of programs and organizations. The achievements and progress have been updated at least annually. Implementation of pollution prevention projects reduced the total amount of waste generated at INEEL in 1997 by approximately 3,100 cubic meters (4,000 cubic yards) (DOE 1998b).

The INEEL waste minimization program has reduced significantly the quantities of hazardous waste generated at INEEL. For example, in 1992, 760 cubic meters (994 cubic yards) of hazardous waste were recycled. Recyclable hazardous materials include metals such as bulk lead, mercury, and chromium; solvents; fuel; and other waste materials (DOE 1995a). Soon the use of nonhazardous chemicals and the recycling of those for which there is no substitute should nearly eliminate the generation of hazardous waste (DOE 1996c).

Another goal of the INEEL waste minimization program is to reduce nonhazardous waste generation by 33 percent by the end of 1999 (DOE 1998b). During 1993 through 1995, INEEL recycled more than 680,400 kilograms (1.5 million pounds) of paper and cardboard (DOE 1999h). Efforts are also underway to expand the recycling program to include asphalt and metals and to convert scrap wood into mulch (DOE 1995a).

3.2.11.9 Preferred Waste Management Alternatives From the Final Waste Management Programmatic EIS and Associated Records of Decision

Preferred waste management alternatives from the *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* (Waste Management Programmatic EIS) (DOE 1997a) are shown in **Table 3–14** for the waste types analyzed in this EIS. Management of this waste could result in the construction of new waste management facilities at INEEL and the closure of other facilities. Decisions on the various waste types were announced in a series of Records of Decision that have been issued on the Waste Management Programmatic EIS. The transuranic waste Record of Decision was issued on January 20, 1998 (63 FR 3629); the hazardous waste Record of Decision on August 5, 1998 (63 FR 41810); the high-level radioactive waste Record of Decision on August 26, 1999 (64 FR 46661); and the low-level and mixed low-level radioactive waste Record of Decision on February 25, 2000 (65 FR 10061). The transuranic waste Record of Decision states, "...each of the Department's sites that currently has or will generate [sic] transuranic waste will prepare and store its transuranic waste on site...." The hazardous waste Record of Decision states that most DOE sites will continue to use offsite facilities for the treatment and disposal of major portions of their nonwastewater hazardous waste, and the Oak Ridge Reservation and SRS will continue to treat some of their own nonwastewater hazardous waste on site in existing facilities, where this is economically favorable. The high-level radioactive waste Record of Decision states that immobilized high-level radioactive waste will be stored at the site of generation. DOE decided in the Record of Decision for the management and disposal of low-level and mixed low-level radioactive waste to perform minimum treatment of low-level radioactive waste at all sites and continue, to the extent practicable, disposal of onsite low-level

radioactive waste at INEEL, Los Alamos National Laboratory, the Oak Ridge Reservation, and SRS. For the management and disposal of mixed low-level radioactive waste, DOE decided to treat this waste at the Hanford site, INEEL, the Oak Ridge Reservation, and SRS, with disposal at the Hanford site and the Nevada Test Site. More detailed information concerning DOE's alternatives for the future configuration of waste management facilities at INEEL is presented in the Waste Management Programmatic EIS and the transuranic, hazardous, high-level, and low-level and mixed low-level radioactive waste Records of Decision.

Table 3–14 Preferred INEEL Waste Management Alternatives From the Waste Management Programmatic EIS and Associated Records of Decision

Waste Type	*Preferred Action*
High-level radioactive	DOE prefers onsite storage of INEEL's immobilized high-level radioactive waste pending disposal in a geologic repository. [a]
Transuranic and mixed transuranic	DOE has decided that INEEL should prepare and store its transuranic waste on site pending disposal at the Waste Isolation Pilot Plant. [b]
Low-level radioactive	DOE has decided to treat INEEL low-level radioactive waste on site. INEEL has been selected as one of the regional disposal sites for low-level radioactive waste. [c]
Mixed	DOE has decided to treat INEEL mixed waste on site, including the possibility of treating mixed waste generated at other sites. INEEL was not selected as one of the regional disposal sites for mixed waste. [c]
Hazardous	DOE has decided to continue to use commercial facilities for treatment of INEEL nonwastewater hazardous waste and onsite facilities for treatment of wastewater hazardous waste. [d]

[a] From the Record of Decision for high-level radioactive waste (64 FR 46661).
[b] From the Record of Decision for transuranic waste (63 FR 3629).
[c] From the Record of Decision for low-level and mixed low-level radioactive waste (65 FR 10061).
[d] From the Record of Decision for hazardous waste (63 FR 41810).
Sources: DOE 1997a, 63 FR 3629, 63 FR 41810.

3.3 SAVANNAH RIVER SITE

SRS is located on about 80,130 hectares (198,000 acres) in southwest South Carolina. The site is approximately 40 kilometers (25 miles) southeast of Augusta, Georgia, and 19 kilometers (12 miles) south of Aiken, South Carolina. First established in 1950, SRS has been involved in tritium operations and nuclear material production for more than 40 years. Today the site includes 16 major production, service, research, and development areas, not all of which are currently in operation. The site is owned by the Federal Government and is administered, managed, and controlled by DOE. It is bordered by the Savannah River to the southwest and includes portions of three South Carolina counties: Aiken, Allendale, and Barnwell.

There are more than 3,000 facilities at SRS, including 740 buildings with 511,000 square meters (5,500,000 square feet) of floor area. Major nuclear facilities at SRS include fuel and plutonium storage facilities; target fabrication facilities; nuclear material production reactors; chemical separation plants; a uranium fuel processing area; liquid high-level radioactive waste tank farms; a waste vitrification facility; and the Savannah River Technology Center. SRS processes nuclear materials into forms suitable for continued safe storage, use, or transportation to other DOE sites. Tritium recycling facilities at SRS empty tritium from expired reservoirs, purify it to eliminate the helium decay product, and fill replacement reservoirs with specification tritium for nuclear stockpile weapons. Filled reservoirs are delivered to Pantex for weapons assembly and directly to the U.S. Department of Defense to replace expired reservoirs. Historically, DOE has produced tritium at SRS, but has not produced any since 1988.

3.3.1 Land Resources

3.3.1.1 Land Use

Forest and agricultural land predominate in the areas bordering SRS (**Figure 3–5)**. There are also significant open water and nonforested wetlands along the Savannah River Valley. Incorporated and industrial areas are the only other significant land uses. There is limited urban and residential development bordering SRS. The closest residences are to the west, north, and northeast within 61 meters (200 feet) of the site boundary. The three counties in which SRS is located, Aiken, Allendale, and Barnwell, have not zoned any of the site land.

Outdoor public recreation facilities are plentiful and varied in the SRS region. Included are the Sumter National Forest, 76 kilometers (47 miles) to the northwest; Santee National Wildlife Refuge, 80 kilometers (50 miles) to the east; and Strom Thurmond Reservoir, 69 kilometers (43 miles) to the northwest. There are also a number of state, county, and local parks in the region, most notably Redcliffe Plantation, Rivers Bridge, Barnwell and Aiken County State Parks in South Carolina, and Mistletoe State Park in Georgia. The Crackerneck Reserve, which occupies 4,047 hectares (10,000 acres) of SRS adjacent to the Savannah River, is open to public use (DOE 1999e).

Land use at SRS can be classified into three major categories: forest/undeveloped, water/wetlands, and developed facilities. Approximately 58,500 hectares (144,600 acres), or 73 percent of the site, is undeveloped. Wetlands, streams, and lakes account for 18,000 hectares (44,500 acres), or 22 percent of the site. Developed facilities, including production and support areas, roads, and utility corridors, encompass 4,000 hectares (9,900 acres), or 5 percent of SRS. Woodland areas are managed primarily for timber production. The U.S. Forest Service, under an interagency agreement with DOE, harvests about 730 hectares (1,800 acres) of timber from SRS each year. In 1972, DOE designated all of SRS as a National Environmental Research Park. The National Environmental Research Park is used by the national scientific community to study the impacts of human activities on the cypress swamp and hardwood forest ecosystems. DOE has set aside approximately 5,700 hectares (14,100 acres) of SRS exclusively for nondestructive environmental research.

Land use in F-Area is classified as heavy industrial. The many facilities located in this area historically have been associated with chemical and physical processes used to separate uranium, plutonium, and fission products (DOE 1996b). Of the many buildings situated in these areas, the F-Canyon is the dominant structure.

Land use in L-Area is classified as heavy industrial. Facilities located in the area historically have been associated with nuclear materials production for national defense. The L-Reactor was shut down in 1988 for safety upgrades and has not restarted (DOE 2000).

3.3.1.2 Visual Resources

The dominant viewshed in the vicinity of SRS consists mainly of agricultural land and forest, with some limited residential and industrial areas. The SRS landscape is characterized by wetlands and forested upland hills. DOE facilities are scattered throughout the site and are brightly lit at night. These facilities are generally not visible off site, as views are limited by rolling terrain, normally hazy atmospheric conditions, and heavy vegetation. The only areas visually impacted by the DOE facilities are those within the view corridors of State Highway 125 and SRS Road 1.

The developed areas and utility corridors (transmission lines and aboveground pipelines) of SRS are consistent with a Visual Resource Management Class IV rating, in which management activities dominate the view and are the focus of viewer attention (DOI 1986). The remainder of SRS generally ranges in Visual Resource Management rating from Class II to Class III. Management activities within these classes may be seen, but should not dominate the view.

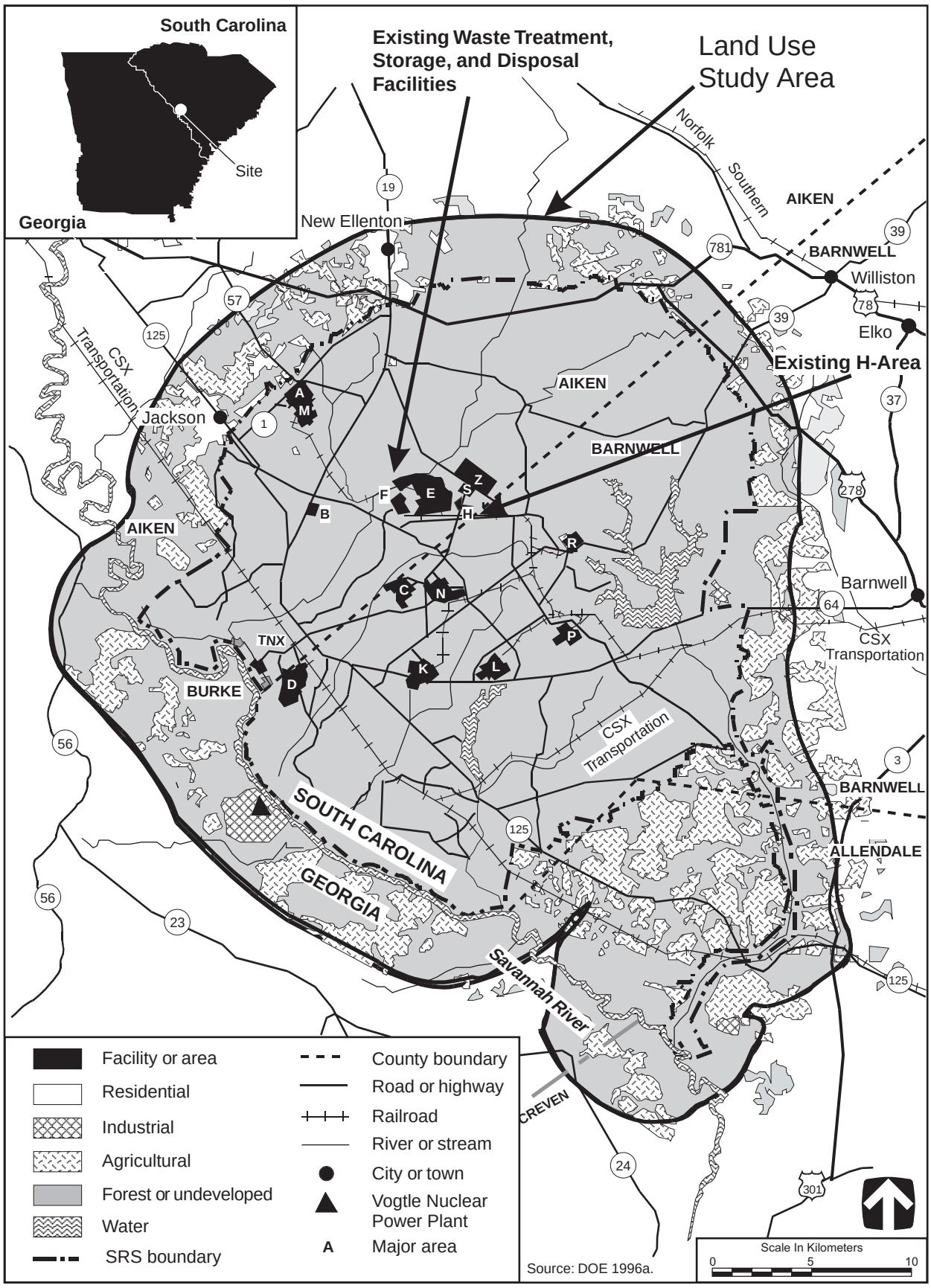

Figure 3–5 Generalized Land Use at SRS and Vicinity

Industrial facilities within F-Area and L-Area consist of large concrete structures, smaller administrative and support buildings, and parking lots. Structures generally range in height from 3 to 30 meters (10 to 100 feet). Facilities in these areas are brightly lit at night and are visible when approached via SRS access roads. However, neither area is visible from State Highway 125 or SRS Road 1 because of the distances involved and the presence of heavily wooded areas next to the roadways. Visual resource conditions in the F-Area and L-Area hold a Visual Resource Management Class IV rating.

3.3.2 Site Infrastructure

Site infrastructure includes those utilities and other resources required to support modification and continued operation of mission-related facilities identified under the various alternative actions. SRS comprises numerous research, processing, and administrative facilities. An extensive infrastructure system supports these facilities, as shown in **Table 3–15.**

Table 3–15 SRS-Wide Infrastructure Characteristics

Resource	*Current Usage*	*Site Capacity*
Transportation		
Roads (kilometers)	230	Not applicable
Railroads (kilometers)	103	Not applicable
Electricity		
Energy consumption (megawatt hours per year)	420,000	5,200,000
Peak load (megawatts)	70	330
Fuel		
Natural gas (cubic meters per year)	Not applicable	Not applicable
Oil (liters per year)	28,400,000	Not applicable [a]
Coal (tons per year)	210,000	Not applicable [a]
Water (liters per year)	1,780,000,000	3,870,000,000

[a] Low supplies can be replenished by truck or rail.
Source: DOE 1999h.

3.3.2.1 Transportation

SRS has an extensive network—230 kilometers (140 miles)—of roads to meet its onsite intrasite transportation requirements. The railroad infrastructure, which consists of 103 kilometers (64 miles) of track, provides deliveries of large volumes of coal and oversized structural components.

3.3.2.2 Electricity

The SRS electrical grid is a 115-kilovolt system in a ring arrangement that supplies power to operating areas, administrative areas, and independent and support function areas. That system includes about 160 kilometers (100 miles) of transmission lines. Power is supplied to the grid by three South Carolina Electric and Gas Company transmission lines. SRS is situated in and draws its power from the Virginia-Carolina Subregion, an electric power pool area that is a part of the Southeastern Electrical Reliability Council. Most of that power comes from offsite coal-fired and nuclear-powered generating plants.

Current site electricity consumption is about 420,000 megawatt hours per year. Site capacity is about 5.2 million megawatt hours per year. The peak load capacity is 330 megawatts; the peak load usage is 70 megawatts.

3.3.2.3 Fuel

Coal and oil are used at SRS primarily to power the steam plants. Steam generation facilities at SRS include coal-fired powerhouses at A-, D-, and H-Areas and two package steam boilers, which use No. 2 fuel oil, in K-Area. Coal is delivered by rail and is stored in coal piles in A-, D- and H-Areas. Oil is delivered by truck to K-Area. The A-Area powerhouse provides process and heating steam for the main administrative area at SRS. The D-Area powerhouse provides most of the steam for the SRS process area. Natural gas is not used at SRS.

3.3.2.4 Water

A new central domestic water system serves the majority of the site. The system includes: three wells and a 17-million-liter (4.5-million-gallon) per day water treatment plant in A-Area; two wells and an 8.3-million-liter (2.2-million-gallon) per day backup water treatment plant in B-Area; three elevated storage tanks; and a 43-kilometer (27-mile) piping loop. This central loop system has an estimated 1,680-liter (444-gallon) per minute excess capacity that could be increased by the installation of an additional elevated storage tank. Process water is provided to individual site areas.

3.3.2.5 Site Safety Services

The SRS fire department operates under a 12-hour rotational shift schedule, with three fire stations. Among the firefighters and officers are members of the SRS Hazardous Materials Response Team and the Rescue Team, who are responsible for rescues of all types. The fire department is supported by a fleet of 20 vehicles, including a specially prepared emergency response step van and trailer for hazardous materials response, and two boats for waterway spill response and control. Inspections are performed periodically according to National Fire Protection Codes and Standards.

3.3.3 Air Quality and Noise

3.3.3.1 Air Quality

Air pollution refers to any substance in the air that could harm human or animal populations, vegetation, or structures, or that unreasonably interferes with the comfortable enjoyment of life and property. Air pollutants are transported, dispersed, or concentrated by meteorological and topographical conditions. Air quality is affected by air pollutant emission characteristics, meteorology, and topography.

The SRS region has a temperate climate with short, mild winters and long, humid summers. Throughout the year, the climate frequently is affected by warm, moist maritime air masses. The average annual temperature at SRS is 17.3 °C (63.2 °F); temperatures vary from an average daily minimum of 0 °C (32 °F) in January to an average daily maximum of 33.2 °C (91.7 °F) in July. The average annual precipitation at SRS is 114 centimeters (45 inches). Precipitation is distributed fairly evenly throughout the year, with the highest in summer and the lowest in autumn. There is no predominant wind direction at SRS. The average annual wind speed at Augusta National Weather Service Station, the nearest National Weather Service Station, is 2.9 meters per second (6.5 miles per hour) (NOAA 1994).

Nonradiological Releases

SRS is near the center of the Augusta-Aiken Interstate Air Quality Control Region No. 53. None of the areas within SRS and its surrounding counties are designated as nonattainment areas with respect to the NAAQS for criteria air pollutants (40 CFR 50). Applicable NAAQS and state ambient air quality standards are presented in **Table 3–16.**

The primary emission sources of criteria air pollutants at SRS are the nine coal-burning boilers and four fuel oil-burning package boilers that produce steam and electricity, diesel engine-powered equipment, the Defense Waste Processing Facility, groundwater air strippers, the consolidated incineration facility, and various other process facilities. Other emissions and sources include fugitive particulates from coal piles and coal processing facilities, vehicles, controlled burning of forestry areas, and temporary emissions from various construction-related activities.

Table 3–16 presents the ambient air concentrations attributable to sources at SRS. These concentrations are based on dispersion modeling using emissions for the year 1998 (DOE 2000). Only those toxic and hazardous air pollutants that would be emitted for any of the alternatives analyzed in this EIS are presented. Concentrations shown in Table 3–16 are in compliance with applicable guidelines and regulations.

Table 3–16 Comparison of Modeled Ambient Air Concentrations at the SRS Boundary From SRS Sources With Most Stringent Applicable Standards or Guidelines

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)* [a]	*SRS Concentration (micrograms per cubic meter)*
Criteria Pollutants			
Carbon monoxide	8 hours	10,000 [b]	6,900
	1 hour	40,000 [b]	10,000
Nitrogen dioxide	Annual	100 [b]	2.6
Ozone	8 hours	157 [c, d]	(e)
PM_{10}	Annual	50 [b]	25
	24 hours (interim)	150 [b]	133
	24 hours (99^{th} percentile over 3 years)	150 [c]	(f)
$PM_{2.5}$	3-year annual	15 [c]	(f)
	24 hours (98^{th} percentile over 3 years)	65 [c]	(f)
Sulfur dioxide	Annual	80 [b]	34
	24 hours	365 [b]	350
	3 hours	1,300 [b]	200
State-Regulated Pollutants			
Gaseous fluoride	30 days	0.8 [g]	0.11
	7 days	1.6 [g]	0.6
	24 hours	2.9 [g]	1.2
	12 hours	3.7 [g]	2.4
Total suspended particulates	Annual	75 [g]	43.3
Hazardous/Toxic Compounds			
1,1,1-Trichloroethane	24 hours	9,550	22
Benzene	24 hours	150	3.9
Beryllium	24 hours	0.01	Less than 0.01
Biphenyl	24 hours	6	0.03
Ethanolamine	24 hours	200	Less than 0.01
Ethyl benzene	24 hours	4,350	0.12
Ethylene glycol	24 hours	650	0.08
Formaldehyde	24 hours	7.5	Less than 0.01
Glycol ethers	24 hours	Not applicable	Less than 0.01
Hexachloronaphthalene	24 hours	1	Less than 0.01
Hexane	24 hours	200	0.07
Manganese	24 hours	25	0.1
Mercury	24 hours	0.25	0.03
Methyl alcohol	24 hours	1,310	0.9
Methyl ethyl ketone	24 hours	14,750	0.99
Methyl isobutyl ketone	24 hours	2,050	0.51

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)* [a]	*SRS Concentration (micrograms per cubic meter)*
Methylene chloride	24 hours	515	1.8
Naphthalene	24 hours	1,250	0.01
Nitric acid	24 hours	125	6.7
Phenol	24 hours	190	0.03
Phosphorous	24 hours	0.5	Less than 0.01
Sodium hydroxide	24 hours	20	0.01
Toluene	24 hours	2,000	1.6
Trichloroethene	24 hours	6,750	1
Vinyl acetate	24 hours	176	0.02
Xylene	24 hours	4,350	3.8

PM_n = Particulate matter less than or equal to *n* microns in diameter.
[a] The more stringent of the Federal and state standards is presented if both exist for the averaging period.
[b] Federal and state standard.
[c] Standard currently under litigation, but will become enforceable during the life of the project.
[d] New NAAQS for ozone (8 hours limit of 0.08 parts per million [171 micro grams/cubic meter]) will become enforceable during the life of the project.
[e] Ambient concentrations of volatile organic compounds, which are precursors to ozone, can be used to provide a highly conservative bounding estimate for ozone but should not be used for explicit assessments of compliance with the ozone standard. Not all the volatile organic compounds emitted will result in the formation of ozone, and there is no method to directly correlate the two quantities. For purposes of estimating ozone concentrations from all SRS operations, no value for total volatile organic compounds is provided since the estimate would be overly conservative.
[f] No data is available with which to assess particulate matter concentrations.
[g] South Carolina state standard.
Sources: DOE 2000, Bickford et al. 1997, South Carolina R.62.5 (Standards 2 and 8), 40 CFR 50, 62 FR 38855, 62 FR 38652.

Data for 1995 from nearby South Carolina monitors at Jackson, Barnwell, and Beech Island (located 30 kilometers [18.6 miles] west of the site) indicate that the NAAQS for particulate matter, sulfur dioxide, and nitrogen dioxide are not exceeded in the area around SRS. Air pollutant measurements at these monitoring locations during 1995 showed: (1) for nitrogen dioxide, an annual average concentration of 9.4 micrograms per cubic meter; (2) for sulfur dioxide, concentrations of 99 micrograms per cubic meter for 3-hour averaging, 24 micrograms per cubic meter for 24-hour averaging, and 5 micrograms per cubic meter for the annual average; (3) for total suspended particulates, an annual average concentration of 37 micrograms per cubic meter; and (4) for PM_{10}, concentrations of 62 micrograms per cubic meter for 24-hour averaging and 19 micrograms per cubic meter for the annual average.

There are no Prevention of Significant Deterioration Class I areas within 100 kilometers (62 miles) of SRS. None of the facilities at SRS have been required to obtain a Prevention of Significant Deterioration permit (DOE 1996c). There are no Prevention of Significant Deterioration increment-consuming sources at SRS.

The meteorological conditions described for SRS are considered representative of F-Area and L-Area. The primary sources of nonradiological air emissions at F-Area and L-Area are diesel generators.

Radiological Releases

The primary sources of radiological air pollutants at SRS, as well as the localized releases associated with "separations" processing (e.g., F-Canyon), are presented in **Table 3–17** (Arnett and Mamatey 1998a). As shown in the table, tritium accounts for most of the total radioactivity released to the atmosphere from SRS operations. During 1997, about 58,000 curies of tritium (both in elemental and oxide forms) were released from SRS.

Table 3–17 Radiological Gaseous and Airborne Emissions at SRS in 1997 (Curies)

Effluent Type	*Radionuclide* [a]	*Reactors*	*Separations* [b]	*Other Facilities at SRS* [c]	*Total* [d]
Gases and vapors	Tritium (oxide)	5,230	33,400	506	39,136
	Tritium (elemental)	—	18,900	—	18,900
	Tritium (total)	5,230	52,300	506	58,036
	Carbon-14	—	0.031	—	0.031
	Krypton-85	—	9,620	—	9,620
	Iodine-129	—	0.0071	1.2×10^{-7}	0.0071
	Iodine-131	—	0.000029	0.00003	0.000059
	Iodine-133	—	—	0.00049	0.00049
Airborne particulates	Cobalt-57	—	2.1×10^{-7}	—	2.1×10^{-7}
	Cobalt-60	—	3.4×10^{-7}	9.1×10^{-7}	1.3×10^{-6}
	Strontium-89, 90	0.0018	0.00022	0.0003	0.0023
	Zirconium-95	—	—	0.000021	0.000021
	Ruthenium-106	—	—	0.07	0.07
	Antimony-125	—	—	5.9×10^{-7}	5.9×10^{-7}
	Cesium-134	—	1.4×10^{-6}	—	1.4×10^{-6}
	Cesium-137	0.00025	0.00042	0.0042	0.0049
	Cerium-144	—	4.2×10^{-6}	6.1×10^{-6}	0.00001
	Europium-154	—	1.5×10^{-7}	6.0×10^{-6}	6.6×10^{-6}
	Europium-155	—	4.9×10^{-6}	1.7×10^{-6}	6.6×10^{-6}
	Uranium-234	—	8.0×10^{-6}	0.000018	0.000027
	Uranium-235	—	6.3×10^{-7}	1.1×10^{-6}	1.8×10^{-6}
	Uranium-236	—	—	4.8×10^{-7}	4.8×10^{-7}
	Uranium-238	—	0.000019	0.000036	0.000056
	Neptunium-239	—	—	2.2×10^{-7}	2.2×10^{-7}
	Plutonium-238	—	0.000033	0.00036	0.00039
	Plutonium-239	0.00029	0.000051	0.000039	0.00038
	Plutonium-240	—	—	1.1×10^{-6}	1.1×10^{-6}
	Plutonium-241	—	—	0.000052	0.000052
	Americium-241	—	0.000014	8.8×10^{-7}	0.000015
	Americium-243	—	—	0.000018	0.000018
	Curium-244	—	0.000025	0.00013	0.00015
Total		**5,230**	**61,920**	**506.1**	**67,656.1**

[a] Release quantities greater than 10^{-7} curies are presented.
[b] Includes F- and H-Canyon, spent fuel storage at receiving basin, waste management, and tritium facilities.
[c] Other facilities include the Savannah River Technology Center, heavy water processing in D-Area, the reactor material area (M-Area), and other unmonitored diffuse and fugitive sources.
[d] Total might differ from sums due to rounding.
Source: Arnett and Mamatey 1998a.

3.3.3.2 Noise

Major noise sources at SRS are primarily in developed or active areas and include various industrial facilities, equipment, and machines. Most industrial facilities at SRS are far enough from the site boundary that noise levels from these sources at the boundary would not be measurable or would be barely distinguishable from background levels. Major noise emission sources outside of these active areas consist primarily of vehicles and rail operations.

An important contributor to noise levels is traffic to and from SRS along access highways through the nearby towns of New Ellenton, Jackson, and Aiken. Noise measurements recorded during 1989 and 1990 along State Route 125 in the town of Jackson, at a point about 15 meters (50 feet) from the roadway, indicate that

the one-hour equivalent sound level from traffic ranged from 48 to 72 decibels A-weighted. The estimated day-night average sound levels along this route were 66 decibels A-weighted for summer and 69 decibels A-weighted for winter. Similarly, noise measurements along State Route 19 in the town of New Ellenton at a point about 15 meters (50 feet) from the roadway indicate that the one-hour equivalent sound level from traffic ranged from 53 to 71 decibels A-weighted. The estimated day-night average sound levels along this route were 68 decibels A-weighted for summer and 67 decibels A-weighted for winter.

No distinguishing noise characteristics at F-Area and L-Area have been identified. These areas are 8 kilometers (5 miles) and 13 kilometers (8 miles) or more from the site boundary, respectively. Thus, contributions to noise levels at the site boundary from these areas are not measurable.

3.3.4 Water Resources

3.3.4.1 Surface Water

The largest river in the area of SRS is the Savannah River, which borders the site on the southwest. Six streams flow through SRS and discharge into the Savannah River: Upper Three Runs, Beaver Dam Creek, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs. Upper Three Runs has two tributaries, Tims Branch and Tinker Creek; Pen Branch has one tributary called Indian Grave Branch; and Steel Creek has one tributary called Meyers Branch **(Figure 3–6)**.

There are two manmade lakes at SRS: L-Lake, which discharges to Steel Creek, and Par Pond, which discharges to Lower Three Runs. Also, up to 350 to 400 Carolina bays (i.e., closed depressions capable of holding water) occur throughout the site. While none of these bays receive direct effluent discharge, some do receive stormwater runoff (DOE 1996c, DOE 2000, WSRC 1997b).

Water historically has been withdrawn from the Savannah River for use mainly as cooling water; some, however, has been used for domestic purposes. SRS currently withdraws about 140 billion liters (37 billion gallons) per year from the river. Most of this water is returned to the river through discharges to various tributaries (DOE 1996c).

The average flow of the Savannah River is 280 cubic meters (10,000 cubic feet) per second. Five large upstream reservoirs, Jocassee, Keowee, Hartwell, Richard B. Russell, and Strom Thurmond, regulate the flow in the Savannah River, thereby lessening the impacts of drought and flooding on users downstream (DOE 1995b).

Several communities in the area use the Savannah River as a source of domestic water. The nearest downstream domestic water intake is the Beaufort-Jasper Water Authority in South Carolina, which withdraws 0.23 cubic meters (8.1 cubic feet) per second to service about 51,000 people. Treated effluent is discharged to the Savannah River from upstream communities and from treatment facilities at SRS. The average annual volume of flow discharged by the sewage treatment facilities at SRS is about 700 million liters (185 million gallons) (DOE 1996c, Barghusen and Feit 1995). The F- and L-Area facilities are not located within a 100-year flood plain; there is no information available concerning 500-year flood plains (WSRC 1995).

A map showing the 100-year flood plain is presented as Figure 3–6. No federally designated Wild and Scenic Rivers occur within the site (Barghusen and Feit 1995).

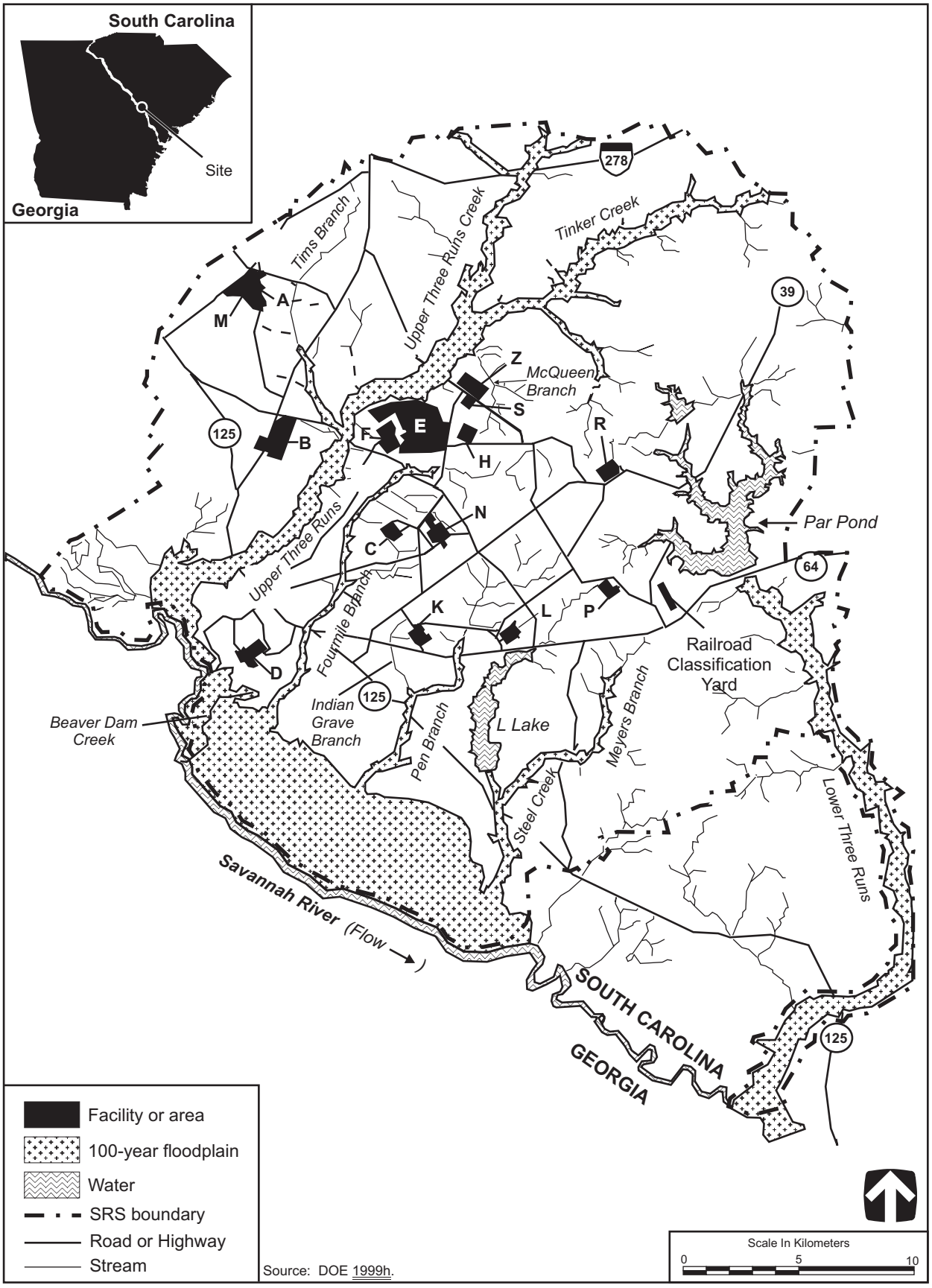

Figure 3–6 Locations of Water Bodies and Flood Plains at SRS

Nonradiological Releases

The Savannah River is classified as a freshwater source that is suitable for primary and secondary contact recreation; drinking, after appropriate treatment; fishing; balanced indigenous aquatic community development and propagation; and industrial and agricultural uses. A comparison of Savannah River water quality upstream (River Mile 160) and downstream (River Mile 120) of SRS showed no significant differences for nonradiological parameters. A comparison of 1997 data shows that the coliform data are within normal fluctuations for river water in this area and the overall cases in which standards were exceeded decreased in number from 1996 (Arnett and Mamatey 1998b). The data for the river's monitoring locations generally meet the freshwater standards set by the State of South Carolina; a comparison of the 1995 and earlier measurements for river samples showed no abnormal deviations.

Surface water rights for SRS are determined by the Doctrine of Riparian Rights, which allows owners of land adjacent to or under the water to use the water beneficially (DOE 1996c). SRS had five National Pollutant Discharge Elimination System (NPDES) permits in 1997, one (SC0000175) for industrial wastewater discharges, one (SCG250162) general permit for utility water discharge, two (SCR000000 and SCR100000) for general stormwater discharges, and one (ND0072125) for land application. Permit SC0000175 regulates 37 outfalls. The 1997 compliance rate for these outfalls was 99.9 percent. The 48 stormwater-only outfalls regulated by the stormwater permits are monitored as required. A pollution prevention plan has been developed to identify where the best available technology and best management practices must be used. For stormwater runoff from construction activities extending over 2 hectares (5 acres), a sediment reduction and erosion plan is required (Arnett and Mamatey 1996, Arnett and Mamatey 1997, Arnett and Mamatey 1998a).

The land around F-Area drains to Upper Three Runs and Fourmile Branch. Upper Three Runs is a large, cool blackwater stream that flows into the Savannah River. It drains about 54,390 hectares (134,400 acres) and has an average discharge of 9.3 cubic meters (330 cubic feet) per second near its mouth. The seven-day, 10-year low flow, which is the lowest flow over any seven days within any 10-year period, is 2.8 cubic meters (100 cubic feet) per second. The stream is about 40 kilometers (25 miles) long, yet only its lower reaches extend through SRS. It receives more water from underground sources than any other SRS stream and, therefore, has lower dissolved solids, hardness, and pH values. It is the only major stream on the site that has not received thermal discharges. It receives permitted discharges from several areas at SRS, including A-, B-, F-, H-, and S-Areas. Flow from the sanitary wastewater discharge averages less than 0.001 cubic meters (0.035 cubic feet) per second. A comparison with the seven-day, 10-year low flow of 2.8 cubic meters (100 cubic feet) per second in Upper Three Runs shows that the present discharges are very small (DOE 1994b, DOE 1995b).

Fourmile Branch is a blackwater stream affected by past operational practices at SRS. Its headwaters are near the center of the site, and it flows southwesterly before discharging into the Savannah River. The watershed is about 5,420 hectares (13,400 acres) and receives permitted effluent discharges from F-Area and H-Area. This stream received cooling water discharges from the C-Reactor while it was operating. Since those discharges ceased in 1985, the maximum recorded temperature in the stream has been 32 °C (90 °F), as opposed to ambient water temperatures that exceeded 60 °C (140 °F) when the reactor was operating. The average flow in the stream during the C-Reactor operation was 11.3 cubic meters (400 cubic feet) per second; since then, flows have averaged 1.8 cubic meters (64 cubic feet) per second. In its lower reaches, this stream widens and flows via braided channels through a delta. Downstream of this delta area, it reforms into one main channel, and most of the flow discharges into the Savannah River at River Mile 152, although a small portion flows west and enters Beaver Dam Creek. When the Savannah River floods, water from Fourmile Branch flows along the northern boundary of the flood plain and joins with other site streams to exit the swamp via Steel Creek instead of flowing directly into the Savannah River (DOE 1995b).

The land around L-Area drains to Steel Creek and Pen Branch. In its headwaters, Pen Branch is a largely undisturbed blackwater stream. Pen Branch and Indian Grave Branch drain an area of about 5,440 hectares (13,440 acres). Pen Branch flows southwesterly from its headwaters east of the K-Area to the Savannah River Swamp. At the swamp it flows parallel to the Savannah River for about 8 kilometers (5 miles) before it enters and mixes with Steel Creek. If the K-Reactor and its cooling tower were to operate, the flow in Indian Grave Branch would be reduced and a large part of its flow would be from cooling tower blowdown. This change would alter the water quality and temperature and flow regimes in Pen Branch. Currently, the Pen Branch system receives nonthermal effluent from K-Area and sanitary effluent from the Central Shops (N-Area). In water year 1991, the mean flow of Pen Branch at SC125 was 4.1 cubic meters (145 cubic feet) per second. Since the shutdown of the K-Reactor, the mean temperature of Pen Branch has been 22 °C (72 °F) and the flow at Road A-13.2 has averaged 0.55 cubic meters (19.3 cubic feet) per second (DOE 1995b; DOE 1997b).

The headwaters of Steel Creek originate near the P-Reactor. The creek flows approximately 3 kilometers (2 miles) before it enters the headwaters of L-Lake. L-Lake is 6.5 kilometers (4 miles) long with an area of about 420 hectares (1,040 acres). Flow from the outfall of L-Lake travels about 5 kilometers (3 miles) before entering Savannah River Swamp and then another 3 kilometers (1.9 miles) before entering the Savannah River. Myers Branch joins Steel Creek downstream of the L-Lake dam. The total area drained by the Steel Creek-Myers Branch system is about 9,070 hectares (22,400 acres). When the L-Reactor was operating, Steel Creek received cooling water from the L-Reactor, ash basin runoff, nonprocess cooling water, powerhouse wastewater, reactor process effluent, sanitary treatment plant effluent, and vehicle wash waters. During water year 1996, the mean flow rate of Steel Creek was 1.7 cubic meters (59.2 cubic feet) per second (DOE 2000).

Radiological Releases

Table 3–18 summarizes the radioactive liquid effluent released at SRS during 1997 (Arnett and Mamatey 1998a). As shown in the table, tritium accounts for most of the radioactivity discharged in SRS liquid effluent. In regard to actinides in nearby streams, trace amounts of uranium and plutonium were detected at a number of stream transport locations. Consequently, these small amounts were incorporated into the source term used for the calculation of the annual dose.

Table 3–18 Radiological Liquid Effluent at SRS in 1997 (Curies)

Radionuclide [a]	*Reactors*	*Separations* [b]	*Other Facilities at SRS* [c]	*Total* [d]
Tritium (oxide)	2,910	5,240	404	8,550
Strontium-89/90 [e]	0.065	0.14	0.0092	0.21
Iodine-129	—	0.078	—	0.078
Cesium-137	0.0029	0.045	—	0.048
Uranium-234	0.0045	0.023	0.00013	0.028
Uranium-235	0.000049	0.00072	3.6×10^{-6}	0.00078
Uranium-238	0.0038	0.026	0.00018	0.03
Plutonium-238	0.000042	0.00096	2.6×10^{-6}	0.001
Plutonium-239 [f]	0.011	0.034	5.6×10^{-6}	0.05
Americium-241	—	7.8×10^{-6}	2.1×10^{-6}	9.9×10^{-6}
Curium-244	—	2.9×10^{-6}	4.1×10^{-7}	3.3×10^{-6}

[a] Release quantities greater than 10^{-7} curies are presented.
[b] Representative of F- and H-Canyon operations, which include separations, waste management, and tritium facilities.
[c] Other facilities include the Savannah River Technology Center, heavy water processing in D-Area, the multipurpose pilot plant campus, and the reactor material area (M-Area).
[d] Totals might differ from sums due to rounding.
[e] Includes other unidentified beta emissions.
[f] Includes other unidentified alpha emissions.
Source: Arnett and Mamatey 1998a.

3.3.4.2 Groundwater

Aquifers are classified by Federal and state authorities according to use and quality. The Federal classifications include Class I, II, and III groundwater. Class I groundwater is either the sole source of drinking water or is ecologically vital. Class IIA and IIB are current or potential sources of drinking water (or other beneficial use), respectively. Class III is not considered a potential source of drinking water and is of limited beneficial use.

Although many different systems have been used to describe groundwater systems at SRS, for this EIS the system used in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c) has been adopted. The uppermost aquifer is referred to as the water table aquifer. It is supported by the leaky "Green Clay" aquitard, which confines the Congaree aquifer. Below the Congaree aquifer is the leaky Ellenton aquitard, which confines the Cretaceous aquifer, also known as the Tuscaloosa aquifer. In general, groundwater in the water table aquifer flows downward to the Congaree aquifer or discharges to nearby streams. Flow in the Congaree aquifer is downward to the Cretaceous aquifer or horizontal to stream discharge or the Savannah River, depending on the location within SRS.

Groundwater in the area is used extensively for domestic and industrial purposes. Most municipal and industrial water supplies in Aiken County are withdrawn from Cretaceous intermediate to deep aquifer units, while small domestic supplies are withdrawn from the water table aquifer. In Barnwell and Allendale counties, the Congaree aquifer supplies some municipal users. It is estimated that about 13 billion liters (3.4 billion gallons) per year are withdrawn from the aquifers within a 16-kilometer (10-mile) radius of the site, which is similar to the volume used by SRS (DOE 1996c). The Cretaceous aquifer is an important water resource for the SRS region. Aiken, South Carolina, for example, uses the Cretaceous aquifer for drinking water. The water is generally soft, slightly acidic, and low in dissolved and suspended solids (DOE 1995b).

Groundwater is the only source of domestic water at SRS (DOE 1995b). All groundwater at SRS is classified by the EPA as a Class II water source, and depth to groundwater ranges from near the surface to about 46 meters (150 feet) (DOE 1996c). SRS withdrawals of groundwater to support site operations range from 34,000 to 45,000 cubic meters (9 to 12 million gallons) per day (DOE 2000). There are no designated sole source aquifers in the area (Barghusen and Feit 1995).

Groundwater ranges in quality across the site. In some areas it meets drinking water quality standards, while in areas near some waste sites it does not. The Cretaceous aquifer is generally unaffected except for an area near A-Area, where trichlorothylene has been reported. Trichlorothylene also has been reported in the A- and M-Areas in the Congaree aquifer. Tritium has been reported in groundwater in the Separations Area. The water table aquifer is contaminated with solvents, metals, and low levels of radionuclides at several SRS sites and facilities. Groundwater eventually discharges into onsite streams or the Savannah River (DOE 1996c), but groundwater contamination has not been detected beyond SRS boundaries (DOE 1995b).

Groundwater rights in South Carolina are associated with the absolute ownership rule. Owners of land overlying a groundwater source are allowed to withdraw as much water as they desire; however, the state requires users who withdraw more than 379,000 liters (100,000 gallons) per day to report their withdrawals. SRS is required to report because its usage is above the reporting level (DOE 1996c).

Groundwater in the shallow, intermediate, and deep aquifers flows in different directions, depending on the depths of the streams that cut the aquifers. The shallow aquifer discharges to Upper Three Runs and Fourmile Branch. Shallow groundwater in the vicinity of F-Area flows toward Upper Three Runs, McQueen Branch, or Fourmile Branch. Groundwater in the intermediate and deep aquifers flows horizontally toward the Savannah River and southeast toward the coast (DOE 1994b).

Groundwater also moves vertically. In the shallow aquifer, it moves downward until its movement is obstructed by impermeable material. Operating under a different set of physical conditions, groundwater in the intermediate and deep aquifers flows mostly horizontally. Near F-Area, it moves upward because of higher water pressure below the confining unit between the upper and lower aquifers. This upward movement helps to protect the lower aquifers from contaminants found in the shallow aquifer. The depth to groundwater in F-Area varies from about 1 to 20 meters (3.3 to 66 feet) (DOE 1994b).

Groundwater quality in F-Area is not significantly different from that for the site as a whole. It is abundant, usually soft, slightly acidic, and low in dissolved solids. High dissolved iron concentrations occur in some aquifers. Where needed, groundwater is treated to raise the pH and remove iron (DOE 1994b).

Groundwater quality in the F-Area can exceed drinking water standards for several contaminants. Near the F-Area seepage basins and inactive process sewer line, radionuclide contamination is widespread. Most of these wells contain tritium above drinking water standards. Other wells exhibit gross alpha, gross beta, strontium-90, and iodine-129 above their standards. Other radionuclides found above proposed standards in several wells include americium-241; curium-243 and -244; radium-226 and -228; strontium-90; total alpha–emitting radium; and uranium-233, -234, -235, and -238. Cesium-137, curium-245 and -246, and plutonium-238 also were found (Arnett and Mamatey 1996).

Near the F-Area Tank Farm, tritium, mercury, nitrate-nitrite as nitrogen, cadmium, gross alpha, and lead were detected above drinking water standards in one or more wells. The pH exceeded the basic standard, and trichlorofluoromethane (freon-11), which has no drinking water standard, was present in elevated levels (Arnett and Mamatey 1996).

At the F-Area Sanitary Sludge Land Application Site, tritium, specific conductance, lead, and copper were found to exceed their drinking water standards in one or more wells. Groundwater near the F-Area Acid/Caustic Basin consistently exceeded drinking water standards for gross alpha. Total alpha-emitting radium, alkalinity, gross beta, nitrate as nitrogen, and pH were above their respective standards in one or more wells. The groundwater near the F-Area Coal Pile Runoff Containment Basin did not exceed any chemical or radiological standard during 1995 (Arnett and Mamatey 1996).

L-Area groundwater exceeds guidelines for tritium, other radionuclides, carbon disulfide, chlorinated and volatile organics, and metals. Groundwater beneath the L-Area Disassembly Basin has been contaminated with metals, chlorinated organics, and tritium (DOE 2000).

3.3.5 Geology and Soils

Coastal Plain sediments beneath SRS overlie a basement complex composed of Paleocene crystalline and Triassic sedimentary formations of the Dunbarton Basin. Small and discontinuous zones of calcareous sand (i.e., sand containing calcium carbonate [calcite]), which potentially is subject to dissolution by water, are beneath some parts of SRS. If dissolution occurs in these zones, potential underground subsidence resulting in settling of the ground surface could occur. No settling as a result of dissolution of these zones has been identified. No economically viable geologic resources have been identified at SRS.

In the immediate region of SRS, there are no known capable faults. A capable fault is one that has had movement at or near the ground surface at least once within the past 35,000 years or recurrent movement within the past 500,000 years. Several faults have been identified from subsurface mapping and seismic surveys within the Paleozoic and Triassic basement beneath SRS. These are shown in Figure 3.1-3 of the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000). The largest fault is the Pen Branch Fault. However, there is no evidence of movement within the last 38 million years along this fault.

Three earthquakes have occurred inside the SRS boundary between 1985 and 1997. The acceleration produced by these earthquakes did not activate seismic monitoring instruments in the reactor areas (these instruments have detection limits of 0.002 g). Existing information does not conclusively correlate these earthquakes with any of the known faults on the site (DOE 1999h). Historically, two large earthquakes have occurred within 160 kilometers (100 miles) of SRS. The Charleston earthquake of 1886 had an estimated Richter magnitude of 6.8, while the Union County, South Carolina, earthquake of 1913 had an estimated Richter magnitude of 6.0. The SRS area experienced an estimated peak horizontal acceleration of 0.10 g during the Charleston earthquake. An earthquake with a maximum horizontal acceleration of 0.2 g is estimated to have an annual probability of occurrence of 1 in 5,000 at SRS. An earthquake of this magnitude would not result in structural damage since this represents the design-basis earthquake (DOE 1995c).

There is no volcanic hazard at SRS. The area has not experienced volcanic activity within the last 230 million years. Future volcanism is not expected because SRS is along the passive continental margin of North America.

The soils at SRS are primarily sands and sandy loams. The somewhat excessively drained soils have a thick, sandy surface layer that extends to a depth of 2 meters (6.6 feet) or more in some areas. Soil units that meet the soil requirements for prime farmland soils exist on SRS. However, the U.S. Department of Agriculture's Natural Resources Conservation Service does not identify these as prime farmlands due to the nature of site use; that is, the lands are not available for the production of food or fiber. The soils at SRS are considered acceptable for standard construction techniques.

The soils of the F-Area and L-Area fall within the Fuquay-Blanton-Dothan Association. This association consists of nearly level to sloping, well-drained soils on broad upland ridges. Soils in this association have moderately thick, sandy surface and subsurface layers and a loamy subsoil (WSRC 1997b). Most soils within the F-Area and L-Area have been disturbed by site development activities.

3.3.6 Ecological Resources

Ecological resources include terrestrial resources, wetlands, aquatic resources, and threatened and endangered species. Material presented in this section, unless otherwise noted, is from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c).

3.3.6.1 Terrestrial Resources

Most of SRS has remained undeveloped since it was established in 1950. Only about 5 percent of the site is occupied by DOE facilities. Five major plant communities have been identified at SRS (**Figure 3-7**). Of these, the largest is the loblolly, longleaf, slash pine community, which covers approximately 65 percent of the site. This community type, as well as upland hardwood-scrub oak, occurs primarily in upland areas. Swamp forests and bottomland hardwood forests are found along the Savannah River and the numerous streams that traverse SRS. More than 1,300 taxa of vascular plants have been identified on the site.

Because of the variety of plant communities on the site, as well as the region's mild climate, SRS supports a diversity and abundance of wildlife, including 44 amphibian, 59 reptile, 255 bird, and 54 mammal species (DOE 1999h). Common species at SRS include the slimy salamander, eastern box turtle, Carolina chickadee, common crow, eastern cottontail, and gray fox. A number of game animals are found on SRS; however, except for the Crackerneck Reserve, only the whitetail deer and feral hog are hunted on site. Raptors, such as the Cooper's hawk and black vulture, and carnivores, such as the gray fox and raccoon, are ecologically important groups on SRS.

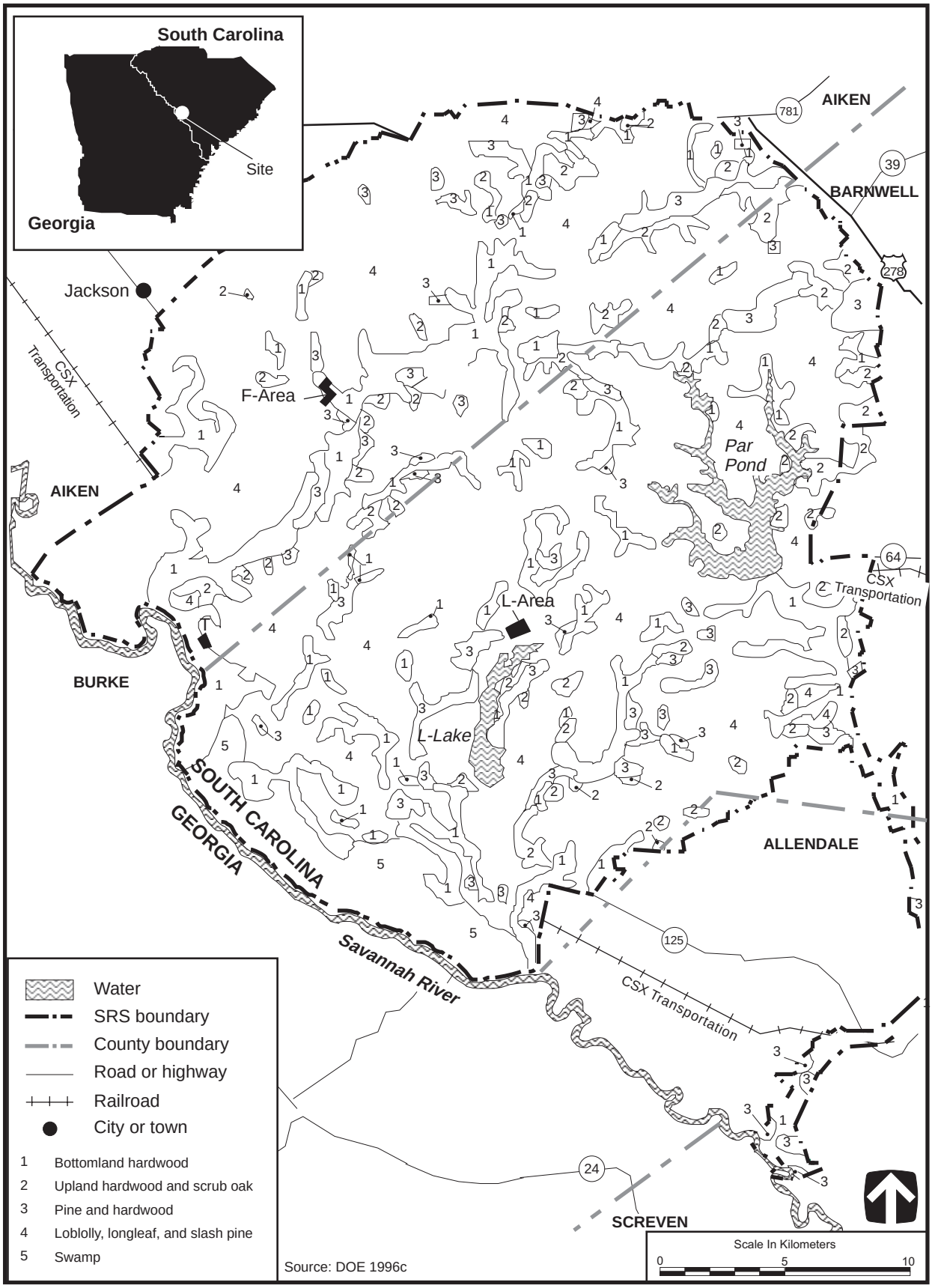

Figure 3–7 Distribution of Plant Communities at SRS

F-Area is an industrial area situated on an upland plateau between the drainage areas of Upper Three Runs and Fourmile Branch. It is surrounded primarily by evergreen forests with areas of grassland, scrub-shrub, and barren land also present. A roughly 6-hectare (15-acre) oak-hickory forest area designated as a National Environmental Research Park set aside is located northwest of the site. Bottomland hardwood forest areas are located along Upper Three Runs and Fourmile Branch. Buildings, paved parking lots, graveled construction areas, and laydown yards dominate this heavily industrialized area; little natural vegetation remains inside the fenced areas (DOE 1996b, DOE 1999h). A total of 41 animal species have been identified in and around F-Area, including 18 species of birds, 11 species of mammals, and 12 species of reptiles (WSRC 1997a).

L-Area is an industrial area largely surrounded by the loblolly, longleaf, and slash pine community, although an area of pine-hardwood community is located to the west. L-Area lies within the Steel Creek drainage just north of L-Lake (Figure 3–7). Plant communities found along Steel Creek include bottomland hardwood. While grassy areas occur within L-Area, it is largely disturbed with little vegetation. A total of 35 animal species have been identified in and around L-Area, including 15 species of birds, 8 species of mammals, and 12 species of reptiles (WSRC 1997a).

3.3.6.2 Wetlands

SRS contains approximately 19,800 hectares (49,000 acres) of wetlands, most of which are associated with flood plains, streams, and impoundments. Wetlands on the site may be divided into the following categories: bottomland hardwoods, cypress-tupelo, scrub-shrub, emergent, and open water. The most extensive wetland type on SRS is swamp forest associated with the Savannah River flood plain, which covers approximately 3,800 hectares (9,390 acres). Past releases of cooling water effluent into site streams and the Savannah River Swamp have resulted in shifts in plant community composition, including reduction in bottomland forests along streams and replacement of bald cypress by scrub-shrub and emergent vegetation in the swamp. As many as 350 to 400 Carolina bays, a type of wetland unique to the southeastern United States, also are found on SRS (DOE 1999c). These natural shallow depressions occur on interstream areas and range from lakes to shallow marshes, herbaceous bogs, shrub bogs, or swamp forests.

Wetlands in the vicinity of F-Area are associated primarily with Upper Three Runs and Fourmile Branch and their tributaries. These wetlands have been classified as bottomland hardwood. Below C-Area, Fourmile Branch was affected by cooling water discharged from the C-Reactor. These releases resulted in shifts in natural vegetation along the lower stream corridor and where it drains into the Savannah River Swamp. Since areas affected by shutdown of the reactor have revegetated, species composition is not the same as it was originally (WSRC 1997b).

Wetlands in the vicinity of L-Area are associated with Pen Branch, Steel Creek, and L-Lake. Prior to the establishment of SRS, wetlands associated with Pen Branch and Steel Creek were classified primarily as bottomland hardwood forest and swamp forest. Past releases of cooling water from the K-, L-, and P-Reactors resulted in shifts in plant community composition from bottomland forests along the stream corridors and cypress-tupelo in the Savannah River Swamp to scrub-shrub and emergent vegetation. Since shutdown of the reactors, some recovery of these areas has occurred; however, new growth has not always included the same species that were present in the original canopy. Wetlands associated with L-Lake include several shoreline zones, including a submersed and floating-leaf zone, emergent zone, and an upper emergent-shrub zone. Efforts have been made to revegetate both Ben Pranch and L-Lake (WSRC 1997b).

3.3.6.3 Aquatic Resources

Aquatic habitat on SRS includes manmade ponds, Carolina bays, reservoirs, and the Savannah River and its tributaries. There are more than 50 manmade impoundments located throughout the site that support populations of bass and sunfish. Fewer than 20 Carolina bays have permanent fish populations. Species

present in these bays include redfin pickerel, mud sunfish, lake chubsucker, and mosquitofish. Par Pond and L-Lake support similar fish populations, including largemouth bass, black crappie, and various species of pan fish. Sport fishing is permitted only within the Crackerneck Reserve. Commercial fishing is not allowed on SRS, although it does take place on the Savannah River. In the past, water intake structures for the C- and K-Reactors and the D-Area powerhouse caused annual estimated entrainment of approximately 10 percent of the fish eggs and larvae passing the intake canals during the spawning season. In addition, estimated impingement losses were approximately 7,600 fish per year.

Streams in the vicinity of F-Area include Upper Three Runs and Fourmile Branch and their tributaries. Fish species present in Upper Three Runs in the vicinity of F-Area include the dusky shiner, yellowfin shiner, redbreast sunfish, and bluegill. It is important as a spawning area for blueback herring and as a seasonal nursery habitat for American shad, striped bass, and other Savannah River species. Fish species present in Fourmile Branch near F-Area include the dusky shiner, creek chubsucker, yellow bullhead, and spotted sunfish. Studies of fish communities in Upper Three Runs and Fourmile Branch indicated that no measurable community-level impacts were associated with contaminants from the F-Area seepage basins (DOE 1996b, DOE 1999h).

Aquatic resources in the vicinity of L-Area are associated with Pen Branch, Steel Creek, and L-Lake. Pen Branch has been affected over the years by the operation and subsequent shutdown of the K-Reactor. During operations, fish populations in warmed portions of the stream were greatly reduced. With the end of reactor operations, a more diverse fish population has recolonized thermal portions of the stream. Steel Creek also has been affected by DOE operations, including the operation and subsequent shutdown of the L-Reactor, operation of the K-Reactor and the eventual diversion of its cooling waters to Par Pond, and the construction of L-Lake. L-Lake has undergone numerous changes in fish populations since it was first formed in 1985. These changes have been associated with colonization of the lake by fish originally in Steel Creek, as well as introduced fish and operation and eventual shutdown of the L-Reactor. Fish species that are common in the lake include largemouth bass, bluegill, redbreast sunfish, and threadfin shad (WSRC 1997b).

3.3.6.4 Threatened and Endangered Species

As shown in Table 3.7.6-1 in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996a), 61 threatened, endangered, and other special status species listed by the Federal Government or the State of South Carolina may be found in the vicinity of SRS. Ten species are federally or state-listed as threatened or endangered (WSRC 1997b). No critical habitat for threatened or endangered species exists on SRS.

No federally listed threatened or endangered species are known to occur in F-Area, although several species may occur in the general vicinity. The American alligator (listed as threatened by virtue of its similarity in appearance to the endangered American crocodile), while fairly abundant on SRS, is uncommon in F-Area. The nearest active bald eagle nest is located along Pen Branch, 8 kilometers (5 miles) southeast of F-Area. Bald eagles are listed as threatened by the U.S. Fish and Wildlife Service and as endangered by South Carolina. Wood storks have been observed 14.5 kilometers (9 miles) from F-Area, near the Fourmile Branch delta. The closest colony of red-cockaded woodpeckers is 12 kilometers (7.5 miles) to the northeast, but suitable forage habitat exists near F-Area (WSRC 1997b). Both wood storks and red-cockaded woodpeckers are federally and state-listed as endangered. The smooth purple coneflower, the only endangered plant species found on SRS, has been found along Burma Road 4.8 kilometers (3 miles) southwest of F-Area. The state-listed rare Oconee azalea has been found on steep slopes adjacent to the Upper Three Runs flood plain just northwest of F-Area (DOE 1995b).

No federally listed threatened or endangered species are known to occur in L-Area, but several species may exist in the general vicinity. The American alligator has been observed in L-Lake and in Steel Creek below

L-Lake. Bald eagles have been observed in the L-Lake area; the nearest bald eagle nest is located on Pen Branch 3.2 kilometers (2 miles) southeast of L-Area. Wood storks have been observed in the Steel Creek delta, located about 9.8 kilometers (6 miles) south of L-Area. The closest colony of red-cockaded woodpeckers to L-Area is located about 8 kilometers (5 miles) to the east-southeast (WSRC 1997b). The nearest colony of the smooth purple coneflower to the site is located about 2.4 kilometers (1.5 miles) to the east near the junction of SRS Roads 9 and B. The Oconee azalea has been identified on the steep slopes adjacent to the Upper Three Runs flood plain about 12 kilometers (7.5 miles) northwest of L-Area (DOE 1995b). Consultation has been conducted with both the U.S. Fish and Wildlife Service and the state.

3.3.7 Cultural and Paleontological Resources

Cultural resources are human imprints on the landscape and are defined and protected by a series of Federal laws, regulations, and guidelines. Field studies conducted over the past two decades by the University of South Carolina's Institute of Archaeology and Anthropology have provided considerable information about the distribution and content of cultural resources at SRS. About 60 percent of SRS has been surveyed, and 858 archaeological (historic and prehistoric) sites have been identified. There are 67 sites considered potentially eligible for listing on the National Register; most of the sites have not been evaluated yet. No SRS nuclear production facilities have been nominated for the National Register, and there are no plans for nominations. Existing SRS facilities lack architectural integrity and do not contribute to the broad historic theme of the Manhattan Project and the production of World War II era nuclear materials.

Cultural resources at SRS are managed under the terms of a programmatic memorandum of agreement among the DOE Savannah River Operations Office, the South Carolina State Historic Preservation Officer, and the Advisory Council on Historic Preservation, dated August 24, 1990. Guidance on the management of cultural resources at SRS is included in the *Archaeological Resources Management Plan of the Savannah River Archaeological Research Program* (SRARP 1989).

3.3.7.1 Prehistoric Resources

Prehistoric resources are physical properties that remain from human activities that predate written records. Prehistoric resources at SRS consist of villages, base camps, limited-activity sites, quarries, and workshops. An extensive archaeological survey program begun at SRS in 1974 includes numerous field studies such as reconnaissance surveys, shovel test transects, and intensive site testing and excavation. There is evidence of more than 800 prehistoric sites, some of which may fall in the vicinity of the proposed facilities. Fewer than 8 percent of these sites have been evaluated for National Register eligibility.

Within F-Area, land areas have been disturbed over the past 46 years by activities associated with construction and operation of the existing facilities. Although no archaeological surveys have been conducted within the boundary of F-Area, no prehistoric cultural materials have been, or are expected to be, identified within this industrial area.

The potential for prehistoric sites in L-Area is limited. The area is in an archaeological site density zone that has the least potential for prehistoric sites of significance (DOE 2000).

3.3.7.2 Historic Resources

Historic resources consist of physical properties that postdate the existence of written records. Types of historic sites include farmsteads, tenant dwellings, mills, plantations and slave quarters, rice farm dikes, dams, cattle pens, ferry locations, towns, churches, schools, cemeteries, commercial building locations, and roads. About 400 historic sites or sites with historic components have been identified within SRS, and some of these may fall within the locations of the proposed facilities. To date, about 10 percent of the historic sites

have been evaluated for National Register eligibility. Most pre-SRS era historic structures were demolished during the initial establishment of SRS in 1950. Two SRS era buildings built in 1951 remain in use. From a Cold War perspective, SRS has been involved in tritium operations and other nuclear material production for more than 40 years; therefore, some existing facilities and engineering records may have significant historical and scientific content.

Within F-Area, land areas have been disturbed over the past 46 years by activities associated with the construction and operation of the existing facilities. Although no surveys have been conducted within the boundary of F-Area, no historic resources are expected to be identified, with the possible exception of surviving facilities and engineering records from the Cold War era.

The Savannah River Archaeological Research Program has not examined any areas in and immediately around Building 105-L. Archaeological resources in the footprint of the building are unlikely to have survived construction, although 1951 aerial photographs show that houses were present in L-Area before the development of SRS in the early 1950s (DOE 2000). Consultation has been conducted with the State Historic Preservation Office.

3.3.7.3 Native American Resources

Native American groups with traditional ties to the area include the Apalachee, Cherokee, Chickasaw, Creek, Shawnee, Westo, and Yuchi. At different times, each of these groups was encouraged by the English to settle in the area to provide protection from the French, Spanish, or other Native American groups. Main villages of both the Cherokee and Creek were located southwest and northwest of SRS, respectively, but both groups may have used the area for hunting and gathering activities. During the early 1800s, most of the remaining Native Americans residing in the region were relocated to the Oklahoma Territory.

Native American resources in the region include remains of villages or townsites, ceremonial lodges, burials, cemeteries, and natural areas containing traditional plants used in religious ceremonies. Literature reviews and consultations with Native American representatives have revealed concerns related to the American Indian Religious Freedom Act within the central Savannah River Valley, including some sensitive Native American resources and several plants traditionally used in ceremonies.

No onsite areas are subject to Native American Treaty Rights. However, five Native American groups, the Yuchi Tribal Organization, the National Council of Muskogee Creek, the Indian Peoples Muskogee Tribal Town Confederacy, the Pee Dee Indian Association, and the Ma Chis Lower Alabama Creek Indian Tribe, have expressed concern over sites and items of religious significance on SRS. DOE routinely notifies these organizations about major planned actions at SRS and asks them to comment on SRS documents prepared in accordance with NEPA.

In 1991, DOE conducted a survey of Native American concerns about religious rights in the central Savannah River Valley (DOE 1991). During this study, three Native American groups, the Yuchi Tribal Organization, the National Council of Muskogee Creek, and the Indian Peoples Muskogee Tribal Town Confederacy, expressed continuing interest in the SRS region with regard to the practice of their traditional religious beliefs. The Yuchi Tribal Organization and the National Council of Muskogee Creek have expressed concerns that several plant species (e.g., redroot, button snakeroot, and American ginseng) traditionally used in Tribal ceremonies could exist on SRS. Redroot and button snakeroot are known to occur on SRS, but are typically found in wet, sandy areas such as evergreen shrub bogs and savannas. Neither species is likely to be found in F-Area or L-Area because of past clearing associated with past development. In addition to those Native American Tribal organizations noted above, consultation has been conducted with the United Keetowah Band, the Pee Dee Indian Association, and the Ma Chris Lower Alabama Creek Indian Tribe.

3.3.7.4 Paleontological Resources

Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geologic age. Paleontological materials from the SRS area date largely from the Eocene Age (54 to 39 million years ago) and include fossil plants, numerous invertebrate fossils, giant oysters, other mollusks, and bryozoa. With the exception of the giant oysters, all other fossils are fairly widespread and common; therefore, the assemblages have low research potential or scientific value.

Paleontological resources have not been recorded in F-Area and their occurrence in L-Area is unlikely.

3.3.8 Socioeconomics

Statistics for employment and economy are presented for the regional economic area which encompasses 15 counties around SRS that are located in Georgia and South Carolina. Statistics for population and housing, community services, and local transportation are presented for the region of influence. The region of influence is a five-county area in which approximately 90 percent of all SRS employees reside (**Table 3–19**). In 1995, SRS employed 16,625 persons (6.5 percent of the 1996 regional economic area civilian labor force).

Table 3–19 Distribution of Employees by Place of Residence in the SRS Region of Influence, 1997

County	*Number of Employees*	*Total Site Employment (Percent)*
Aiken	8,966	53.9
Columbia	2,209	13.3
Richmond	2,204	13.3
Barnwell	1,112	6.7
Edgefield	242	1.5
Region of influence total	**14,733**	**88.6** [a]

[a] Total differs due to rounding.
Source: HNUS 1997.

3.3.8.1 Regional Economy Characteristics

Between 1990 and 1996, the civilian labor force in the regional economic area increased 3.6 percent to 257,101. In 1996, the unemployment rate in the regional economic area was 7.6 percent, which was greater than the unemployment rate of 6 percent for both Georgia and South Carolina.

In 1995, manufacturing represented the largest sector of employment in the regional economic area (25.6 percent). This was followed by government (20.9 percent) and service activities (19.9 percent). The total for these employment sectors in Georgia was 17.5 percent, 16.8 percent, and 23 percent, respectively. The total for these employment sectors in South Carolina was 23.3 percent, 17.3 percent, and 20.5 percent, respectively.

3.3.8.2 Population and Housing

In 1996, the region of influence estimated population totaled 453,778. Between 1990 to 1996, the region of influence population increased by 8.6 percent, compared with a 13 percent increase in Georgia's population and a 5.7 percent increase in South Carolina's population. Between 1980 and 1990, the number of housing units in the region of influence increased by 25.1 percent, compared with a 30.1 percent increase in Georgia

and a 23.5 percent increase in South Carolina. The total number of housing units within the region of influence for 1990 was 165,443 (DOE 1999h). In 1995, the total number of owner and renter housing units within the region of influence was 171,400 (DOE 1996c). The 1990 homeowner vacancy rate for the region of influence was 2.2 percent, compared with statewide rates of 2.5 percent for Georgia and 1.7 percent for South Carolina. The renter vacancy rate for the region of influence was 10 percent compared with the statewide rates of 12.2 percent for Georgia and 11.5 percent for South Carolina.

3.3.8.3 Community Services

Community services include public education and public safety. In 1997, school districts providing public education in the region of influence were operating at capacities of between 85 to 100 percent. Total student enrollment in the region of influence in 1997 was approximately 89,000, and the student-to-teacher ratio averaged 17 to 1. In 1990, the average student-to-teacher ratios were 10.8 to 1 for Georgia and 11.5 to 1 for South Carolina. In 1997, a total of 973 sworn police officers were serving the five-county region of influence. The average region of influence officer-to-population ratio was 2.1 officers per 1,000 persons. This compares with the 1990 state averages of 2 officers per 1,000 persons for Georgia and 1.8 officers per 1,000 persons for South Carolina.

3.3.8.4 Local Transportation

Vehicular access to SRS is provided by South Carolina State Routes 19, 64, and 125 (Figure 3–5). There is no public transportation to SRS. Rail service in the region of influence is provided by the Norfolk Southern Corporation and CSX Transportation. SRS is provided rail access via Robbins Station on the CSX Transportation line. Waterborne transportation is available via the Savannah River. SRS has no commercial docking facilities, but it has a boat ramp that has accepted large transport barge shipments. Columbia Metropolitan Airport in Columbia, South Carolina, and Bush Field in Augusta, Georgia, receive jet air passenger and cargo service from both national and local carriers.

3.3.9 Environmental Justice

Under Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, Federal agencies are responsible for identifying and addressing the possibility of disproportionately high and adverse health or environmental effects of programs and policies on minority or low-income populations in potentially affected areas. Minority populations refer to all people of color, exclusive of white non-Hispanics. Low-income populations refer to households whose incomes are below the Federal poverty thresholds. In the case of SRS, the potentially affected area includes parts of Georgia and South Carolina.

Data obtained during the 1990 census show that the percentage of minorities for the contiguous United States was 24.1, and the percentages for the States of Georgia and South Carolina were 29.8 and 31.4, respectively. The same census data also show that, of the total population of the contiguous United States, 13.1 percent reported incomes below the poverty threshold, and Georgia and South Carolina reported 14.7 and 15.4 percent, respectively.

The potentially affected area surrounding F-Area is defined by a circle with an 80-kilometer (50-mile) radius centered at Building 221–F (latitude 33°17'11" N, longitude 81°40'38" W). The total population residing within that area in 1990 was 615,734. The proportion of the population around this building that was considered minority was 37.9 percent. At the time of the 1990 census, Blacks were the largest minority group within the potentially affected area, constituting 35.7 percent of the total population. Hispanics constituted about 1 percent, and Asians about 1 percent. Native Americans constituted about 0.2 percent of the population (DOC 1992).

A breakdown of incomes in the potentially affected area is also available from the 1990 census data (DOC 1992). At that time, the poverty threshold was $9,981 for a family of three with one related child under 18 years of age. A total of 107,479 persons (18 percent of the total population) residing within the potentially affected area around F-Area reported incomes below the poverty threshold.

The potentially affected area surrounding L-Area is defined by a circle with a radius equal to 80 kilometers (50 miles) centered at Building 105-L (latitude 33°12'38.5" N and longitude 81°37'26.5" W). The total population residing within the potentially affected area in 1990 was 606,819 persons. Approximately 39.1 percent of the population in 1990 was composed of individuals who identified themselves as having racial or ethnic origins that are used by the Council on Environmental Quality to define minority populations (CEQ 1997). At the time of the 1990 census, Blacks were the largest minority group within the potentially affected area, constituting approximately 36.8 percent of the total population. Less that 3 percent of the total population in the potentially affected area designated themselves as Asian, Native American, or Hispanic (DOC 1992).

Within the potentially affected area in 1990, 107,468 persons (nearly 21 percent of the total population) reported incomes that were less than the threshold for poverty.

3.3.10 Existing Human Health Risk

Public and occupational health and safety issues include the determination of potentially adverse effects on human health that result from acute and chronic exposures to ionizing radiation and hazardous chemicals.

3.3.10.1 Radiation Exposure and Risk

Major sources and levels of background radiation exposure to individuals in the vicinity of SRS are shown in **Table 3–20.** Annual background radiation doses to individuals are expected to remain constant over time. The total dose to the population, in terms of person-rem, changes as the population size changes. Background radiation doses are unrelated to SRS operations.

Table 3–20 Sources of Radiation Exposure to Individuals in the SRS Vicinity Unrelated to SRS Operations

Source	*Effective Dose Equivalent (millirem per year)*
Natural Background Radiation [a]	
Cosmic radiation	27
External terrestrial radiation	28
Internal terrestrial/cosmogenic radiation	40
Radon in homes (inhaled)	200 [b]
Other Background Radiation [c]	
Diagnostic x-rays and nuclear medicine	53
Weapons test fallout	less than 1
Air travel	1
Consumer and industrial products	10
Total	360

[a] Arnett and Mamatey 1998a.
[b] An average for the United States.
[c] NCRP 1987.

Releases of radionuclides to the environment from SRS operations provide another source of radiation exposure to individuals in the vicinity of SRS. Types and quantities of radionuclides released from SRS operations in 1997 are listed in the *Savannah River Site Environmental Report for 1997* (Arnett and Mamatey 1998a). The doses to the public resulting from these releases are presented in **Table 3–21.** These doses fall within radiological limits per DOE Order 5400.5, *Radiation Protection of the Public and Environment*, and are much lower than those of background radiation.

Table 3–21 Radiation Doses to the Public From Normal SRS Operations in 1997 (Total Effective Dose Equivalent)

	Atmospheric Releases		*Liquid Releases*		*Total*	
Members of the Public	***Standard*** [a]	***Actual***	***Standard*** [a]	***Actual*** [b]	***Standard*** [a]	***Actual***
Maximally exposed offsite individual (millirem)	10	0.050	4	0.13	100	0.18
Population within 80 kilometers (50 miles) (person-rem) [c]	None	5.5 [d]	None	2.4	100	7.9
Average individual within 80 kilometers (50 miles) (millirem) [e]	None	0.0089	None	0.0035	None	0.013

[a] The standards for individuals are given in DOE Order 5400.5. As discussed in that Order, the 10-millirem-per-year limit from airborne emissions is required by the Clean Air Act (40 CFR 61), and the 4-millirem-per-year limit is required by the Safe Drinking Water Act. For this EIS, the 4-millirem-per-year value is assumed conservatively to be the limit for the sum of doses from all liquid pathways. The total dose of 100 millirem per year is the limit from all pathways combined. The 100 person-rem value for the population is given in proposed 10 CFR 834, as published in 58 FR 16268. If the potential total dose exceeds the 100 person-rem value, the contractor operating the facility is required to notify DOE.

[b] Conservatively includes all water pathways, not just the drinking water pathway. The population dose includes contributions to Savannah River users downstream of SRS to the Atlantic Ocean.

[c] About 620,100 in 1997. For liquid releases, an additional 70,000 water users in Port Wentworth, Georgia, and Beaufort, South Carolina (about 160 kilometers [98 miles] downstream), are included in the assessment.

[d] This corresponds to the value calculated for Clean Air Act (40 CFR 61) compliance and is consistent with the assumptions used in dose calculations presented in this EIS.

[e] Obtained by dividing the population dose by the number of people living within 80 kilometers (50 miles) of the site for atmospheric releases; for liquid releases, the number of people includes water users who live more than 80 kilometers (50 miles) downstream of the site.

Source: Arnett and Mamatey 1998a.

Using a risk estimator of 500 latent cancer deaths per 1 million person-rem to the public (see Appendix E), the fatal cancer risk to the maximally exposed offsite individual resulting from radiological releases from SRS operations in 1997 is estimated to be 9.0×10^{-8}. That is, the estimated probability of this person dying of cancer at some point in the future from radiation exposure associated with one year of SRS operations is less than 1 in 10 million (it takes several to many years from the time of radiation exposure for a cancer to manifest itself).

According to the same risk estimator, 0.004 excess latent fatal cancers are projected in the population living within 80 kilometers (50 miles) of SRS from normal operations in 1997. To place this number in perspective, it may be compared with the number of fatal cancers expected in the same population from all causes. The 1995 mortality rate associated with cancer for the entire U.S. population was 0.2 percent per year. Based on this mortality rate, the number of latent fatal cancers expected during 1997 from all causes in the population living within 80 kilometers (50 miles) of SRS was 1,240. This expected number of fatal cancers is much higher than the 0.004 latent fatal cancers estimated from SRS operations in 1997.

SRS workers receive the same dose as the general public from background radiation, but they also receive an additional dose from working in facilities with nuclear materials. The average dose to the individual

worker and the cumulative dose to all workers at SRS from operations in 1997 are presented in **Table 3–22.** These doses fall within the radiological regulatory limits of 10 CFR 835. According to a risk estimator of 400 latent fatal cancers per 1 million person-rem among workers (Appendix E), the number of projected fatal cancers among SRS workers from normal operations in 1997 is 0.066. The risk estimator for workers is lower than the estimator for the public because of the absence from the work force of the more radiosensitive infant and child age groups.

Table 3–22 Radiation Doses to Workers From Normal SRS Operations in 1997 (Total Effective Dose Equivalent)

Occupational Personnel	*Onsite Releases and Direct Radiation*	
	Standard [a]	*Actual*
Average radiation worker (millirem)	None [b]	50
Total workers (person-rem)[c]	None	165

[a] The radiological limit for an individual worker is 5,000 millirem per year. However, DOE's goal is to maintain radiological exposure as low as is reasonably achievable. Therefore, DOE has established an administrative control level of 2,000 millirem per year (DOE Order N 441.1); the site must make reasonable attempts to maintain individual worker doses below this level.

[b] No standard is specified for an "average radiation worker;" however, the maximum dose that this worker may receive is limited to that given in footnote "a."

[c] In 1997, 3,327 workers with measurable doses.

Sources: DOE 1995a, DOE 1998d.

A more detailed presentation of the radiation environment, including background exposures and radiological releases and doses, is presented in the *Savannah River Site Environmental Report for 1997* (Arnett and Mamatey 1998a). The concentrations of radioactivity in various environmental media (including air, water, and soil) in the site region (on and off the site) also are presented in that report.

External radiation doses and concentrations of gross alpha, plutonium, and americium in air have been measured in F-Area. Onsite doses are measured for comparison against natural background levels, which are measured at offsite locations; the numerical difference in these measurements may be directly attributable to radiological sources that are located in the vicinity of the onsite measurement location(s). In 1997, the annual dose in F-Area was 105 millirem. This is about 20 millirem higher than the average dose measured at offsite locations. In the same year, the concentration of gross alpha was about 0.0011 picocuries per cubic meter in F-Area, compared with the approximately 0.00099 picocuries per cubic meter measured at the offsite control location. No plutonium-239 was detected in F-Area. Offsite controls also did not detect any plutonium-239 in the air in 1997 (Arnett and Mamatey 1998b).

External radiation doses have been measured in L-Area. In 1997, the annual dose in L-Area was 80 millirem (Arnett and Mamatey 1998b).

3.3.10.2 Chemical Environment

Table 3-16 identifies the hazardous (i.e., carcinogenic and toxic/noncarcinogenic) chemicals that are emitted to the air at SRS. The list includes only those chemicals that have ambient air quality standards and would be emitted under the alternatives analyzed at SRS. This list includes 24 chemicals, including benzene, ethyl benzene, formaldehyde, hexane, manganese, methyl-ethyl-ketone, methylene chloride, naphthalene, toluene, and vinyl acetate (see Table 3-17 for the complete list).

Health impacts on the public may occur by inhaling air containing hazardous chemicals, ingesting contaminated drinking water or food, and direct exposure (skin contact). The primary health impacts from

exposure to hazardous chemicals are from inhalation. Two major health effects are observed from the listed chemicals, the carcinogenic effect and the noncarcinogenic effect. These are presented below.

Carcinogenic Effects: These effects are estimated as the incremental probability of an individual developing cancer over a lifetime as a result of exposure to the potential carcinogen. This could be an incremental or excess individual lifetime cancer risk.

Noncarcinogenic Effects: These effects are determined by the ratio between the calculated or measured concentration of the chemical in the air and the reference concentration or dose. This ratio is known as the hazard quotient. Hazard quotients for noncarcinogens are summed to obtain the hazard index. If the hazard quotient is less than 1, then no adverse health effects are expected.

For some chemicals where the weight of evidence is weak and carcinogenecity is not well established, the impacts of both cancer and noncancer effects were determined. **Table 3-23** summarizes the baseline hazardous chemical impacts to the public. This table lists only those chemicals for which reference concentrations for cancer or toxicity are available from the Integrated Risk Information System. The baseline concentrations are estimates of the highest existing concentrations and represent the highest concentrations to which individuals from the public could be exposed under normal operations (excluding accident conditions). These concentrations are in compliance with applicable guidelines. Additional information on estimating the health impacts of hazardous chemicals is presented in Appendix E, Section E.5.

The exposure of workers to hazardous chemicals varies among facilities and the operational activities, and the available information is insufficient for a meaningful estimate of impacts. Workers are protected by adherence to the OSHA and EPA standards that regulate workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. Monitoring the frequency and amount of chemicals released in operational processes ensures that these standards are not exceeded. Further, DOE requires that the environment in the workplace be as free as possible from recognized hazards that cause, or are likely to cause, illness or physical harm. Therefore, workplace conditions at SRS are substantially better than required by standards.

Table 3–23 Hazardous Chemical Impacts to the Public From Existing Activities at SRS

Chemical	*Annual Concentration (milligrams per cubic meter)*	*Reference Concentration Inhalation (milligrams per cubic meter)*	*Unit Cancer Risk (risk per milligram per cubic meter*)	*Hazard Quotient*	*Cancer Risk*
Benzene	0.0039	None	0.0078	None	0.00003
Ethyl benzene	0.000015	1	None	0.000015	None
Formaldehyde	1.3×10^{-6}	None	0.013	None	1.6×10^{-8}
Hexane	8.75×10^{-6}	0.2	None	0.000044	None
Manganese	0.000013	0.00005	None	0.25	None
Methyl-ethyl-ketone	0.00012	1	None	0.00012	None
Methylene chloride	0.00023	None	0.00047	None	1.1×10^{-7}
Naphthalene	1.3×10^{-6}	0.003	None	0.00042	None
Toluene	0.0002	0.4	None	0.0005	None
Vinyl acetate	2.5×10^{-6}	0.2	None	0.000013	None
Hazard Index				0.25	Not applicable

Sources: EPA 1999, Bickford et al. 1997.

3.3.10.3 Health Effects Studies

One epidemiological study on the general population in communities surrounding SRS has been conducted and published. No evidence of excess cancer mortality, congenital anomalies, birth defects, early infancy deaths, strokes, or cardiovascular deaths was reported. The epidemiological literature on the facility reflects an excess of leukemia deaths among hourly workers; no other health effects for workers are reported. For a more detailed description of the studies reviewed and their findings, and for a discussion of the epidemiologic surveillance program implemented by DOE to monitor the health of current SRS workers, refer to Appendix M.4.7 of the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c).

3.3.10.4 Accident History

Between 1974 and 1988, there were 13 inadvertent tritium releases from the SRS tritium facilities. These releases were attributed to aging equipment in the tritium processing facility and are one of the reasons for the construction of the Replacement Tritium Facility at SRS. A detailed description and study of these incidents and their consequences for the offsite population have been documented by SRS. The most significant were in 1981, 1984, and 1985, when respectively 32,934, 43,800, and 19,403 curies of tritiated water vapor were released. From 1989 through 1992, there were 20 inadvertent releases, all with little or no offsite dose consequences. The largest of the recent releases occurred in 1992 when 12,000 curies of tritium were released.

3.3.10.5 Emergency Preparedness

Each DOE site has established an emergency management program that would be activated in the event of an accident. This program has been developed and maintained to ensure adequate response to most accident conditions and to provide response efforts for accidents not specifically considered. The emergency management program includes emergency planning, preparedness, and response.

The Emergency Preparedness Facility at SRS provides overall direction and control for onsite responses to emergencies and coordinates with Federal, state, and local agencies and officials on the technical aspects of the emergency. Emergency plans have been prepared for specific areas at SRS. Participating government agencies whose plans are interrelated with the SRS emergency plan for action include the States of South Carolina and Georgia, the City of Aiken, and the various counties in the general region of the site. Emergency response support, including firefighting and medical assistance, would be provided by these jurisdictions.

In addition, DOE has specified actions to be taken at all DOE sites to implement lessons learned from the emergency response to an accidental explosion at Hanford in May 1997.

3.3.11 Waste Management

Waste management includes minimization, characterization, treatment, storage, transportation, and disposal of waste generated from ongoing DOE activities. The waste is managed according to appropriate treatment, storage, and disposal technologies, and in compliance with all applicable Federal and state statutes and DOE Orders.

3.3.11.1 Waste Inventories and Activities

SRS manages the following types of waste: high-level radioactive, transuranic, mixed transuranic, low-level radioactive, mixed, hazardous, and nonhazardous. Waste generation rates and the inventory of stored waste from activities at SRS are provided in **Table 3–24**. **Table 3–25** summarizes the SRS waste management capabilities. More detailed descriptions of the waste management system capabilities at SRS are included in

Table 3–24 Waste Generation Rates and Inventories at SRS

Waste Type	*Generation Rate (cubic meters per year)*	*Inventory (cubic meters)*
High-Level Radioactive	1,561	131,000
Transuranic [a]		
Contact-handled	427	6,977
Remotely handled	4	0
Low-Level Radioactive	10,043	1,616
Mixed		
RCRA	1,135	6,940
Toxic Substances Control Act	0	110
Hazardous	74	1,416
Nonhazardous		
Liquid	416,100	Not applicable [b]
Solid	6,670	Not applicable [b]

[a] Includes mixed transuranic waste.
[b] Generally, nonhazardous waste is not held in long-term storage.
Sources: DOE 1999h, except high-level radioactive waste generations rates (DOE 1996c) and high-level radioactive waste inventory (DOE 1997a).

Table 3–25 Waste Management Capabilities at SRS

Facility Name/Description	*Capacity*	*Status*	*Applicable Waste Type*						
			HLW	*TRU*	*Mixed TRU*	*LLW*	*Mixed*	*Haz*	*Non-Haz*
Treatment Facility (cubic meters per year)									
Savannah River Technology Center Ion Exchangers, Evaporators	53,700	On-line	X						
Transuranic Waste Characterization/ Certification Facility	1,720	Planned for 2007		X	X				
Consolidated Incineration Facility and Ashcrete Stabilization Facility	4,630 liquid 17,830 solid	On-line				X	X	X	
F- and H-Area Effluent Treatment Facility	1,930,000	On-line				X	X		
M-, L-, and H-Area Compactors	3,983	On-line				X			
Non-Alpha Vitrification Facility	3,090	Planned				X	X	X	
M-Area Liquid Effluent Treatment Facility	999,000	On-line					X		
M-Area Vendor Treatment Facility	2,470	Planned					X		
Savannah River Technology Center Ion Exchange Treatment Probe	11,200	On-line					X		
Area Supercompactor	5,700	Planned				X			
Z-Area Saltstone Facility	28,400	On-line					X		
Central Sanitary Wastewater Treatment Facility	1,030,000	On-line							X

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed	Haz	Non-Haz
Storage Facility (cubic meters)									
Transuranic Storage Pads	34,400	On-line		X	X				
Defense Waste Processing Facility Organic Waste Storage Tank	568	On-line					X		
Liquid Waste Solvent Tanks	454	Planned					X		
M-Area Process Waste Interim Treatment/Storage Facility	8,300	On-line					X		
Mixed Waste Storage Facilities (645-2N, -295, -43E)	1,905	On-line					X		
Savannah River Technology Center Mixed Waste Storage Tanks	198	On-line					X		
Long-Lived Waste Storage Building	1,064	Planned				X			
Solid Waste Storage Pads	2,657	On-line					X	X	
Buildings 316-M, 710-B, 645-N, and 645-4N	2,515	On-line					X	X	
M-Area Storage Pad	2,160	On-line					X		
F- and H-Area Tank Farm	133,000	On-line	X						
Defense Waste Processing Facility	2,286 canisters	On-line	X						
Disposal Facility (cubic meters)									
Intermediate-Level Radioactive Waste Vaults	3,665	On-line				X			
Low-Activity Waste Vaults	30,500	On-line				X			
Low-Level Radioactive Waste Disposal Facility Slit Trenches	26,000	Planned				X			
Z-Area Saltstone Vaults	1,110,000	On-line				X			

DWPF = Defense Waste Processing Facility, Haz = hazardous, HLW = high-level radioactive waste, LLW = low-level radioactive waste, TRU = transuranic
Sources: DOE 1999h, except high-level radioactive waste (DOE 1996c).

the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c) and the *Savannah River Site Waste Management Final Environmental Impact Statement* (DOE 1995b).

The EPA placed SRS on the National Priorities List in December 1989. In accordance with CERCLA, DOE entered into a Federal Facilities Compliance Agreement with the EPA and the State of South Carolina to coordinate cleanup activities at SRS under one comprehensive strategy. As stated in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c), this Agreement combines the RCRA Facility Investigation Program Plan with a CERCLA cleanup program titled the *RCRA Facility Investigation/Remedial Investigation Program Plan*. More information on regulatory requirements for waste disposal is provided in Chapter 5.

3.3.11.2 High-Level Radioactive Waste

Liquid high-level radioactive waste at SRS is made up of many waste streams generated during the recovery and purification of transuranic waste products and unburned fissile material from spent reactor fuel elements. This waste is separated by waste form, radionuclide, and heat content before their transfer to underground

storage tanks in the F- and H-Area Tank Farms. Processes routinely used to treat liquid high-level radioactive waste are separation, evaporation, and ion exchange. Evaporation produces a cesium-contaminated condensate. Cesium is removed from the condensate, resulting in a low-level radioactive waste stream that is treated in the Effluent Treatment Facility. The remaining high-level radioactive waste stream salts are precipitated; some can be decontaminated. The decontaminated salt solution is sent with residues from the Effluent Treatment Facility to the Defense Waste Processing Z-Area Saltstone Facility, where it is mixed with a blend of cement, fly ash, and blast furnace slag to form grout. The grout is pumped into disposal vaults where it hardens for permanent disposal as solid low-level radioactive waste. The remaining high-level radioactive salt and sludge are immobilized permanently as a glass solid cast in stainless steel containers at the Defense Waste Processing Facility Vitrification Plant. The stainless steel containers are decontaminated to U.S. Department of Transportation standards, welded closed, and temporarily stored on site for eventual transport to and disposal in a repository. Future high-level radioactive waste generation could result from the processing and stabilization of spent nuclear fuel for long-term storage as a result of the Record of Decision (60 FR 28680) on the Programmatic Spent Nuclear Fuel EIS (DOE 1995a), and from remediation or materials recovery activities performed in the F- and H-Canyons.

3.3.11.3 Transuranic Waste

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 soil. Transuranic waste generated since 1986 is stored on 13 concrete pads that are not covered with soil. The transuranic waste storage pads are in the Low-Level Radioactive Waste Disposal Facility (DOE 1999h).

A planned Transuranic Waste Characterization and Certification Facility would provide extensive containerized waste certification capabilities. The facility is needed to prepare transuranic waste for treatment and to certify transuranic waste for disposal at the Waste Isolation Pilot Plant. Drums that are certified for shipment to the Waste Isolation Pilot Plant would be placed in interim storage on concrete pads in E-Area. Low-level radioactive waste containing concentrations of transuranic nuclides between 10 and 100 nanocuries (referred to as alpha-contaminated low-level radioactive waste) is managed like transuranic waste because its physical and chemical properties are similar and similar procedures would be used to determine its final disposition (DOE 1996c). The Waste Isolation Pilot Plant is expected to begin receiving waste from SRS in 2000 (DOE 1999b).

3.3.11.4 Low-Level Radioactive Waste

Both liquid and solid low-level radioactive waste are treated at SRS. Most aqueous low-level radioactive waste streams are sent to the F- and H-Area Effluent Treatment Facility and treated by filtration, reverse osmosis, and ion exchange to remove the radionuclide contaminants. After treatment, the effluent is discharged to Upper Three Runs. The treatment residuals are concentrated by evaporation and stored in the H-Area Tank Farm for eventual treatment in the Z-Area Saltstone Facility. In that facility, waste is immobilized with grout for onsite disposal (DOE 1996c).

After completion of a series of extensive readiness tests, the Consolidated Incinerator Facility began radioactive operations in 1997. The Consolidated Incinerator Facility is designed to incinerate both solid and liquid low-level radioactive waste, mixed waste, and hazardous waste (DOE 1999h).

Solid low-level radioactive waste is segregated into several categories to facilitate proper treatment, storage, and disposal. Solid low-level radioactive waste with a dose rate of less than 200 millirem per hour at 5 centimeters (2 inches) from an unshielded container is considered low-activity waste. If its dose rate is greater than 200 millirem per hour at 5 centimeters (2 inches), it is considered intermediate-activity waste. Intermediate-activity tritium waste is intermediate-activity waste with more than 10 curies of tritium per

container. Long-lived radioactive waste is contaminated with long-lived isotopes that exceed the waste acceptance criteria for onsite disposal (DOE 1996c).

Four basic types of vaults and buildings are used for storing the different waste categories: low-activity radioactive waste vaults, intermediate-level radioactive nontritium vaults, intermediate-level radioactive tritium vaults, and the long-lived radioactive waste storage building. The vaults are below-grade concrete structures, and the storage building is a metal building on a concrete pad (DOE 1996c).

Currently, DOE places low-activity low-level radioactive waste in carbon steel boxes and deposits them in the low-activity waste vaults in E-Area. Intermediate-activity low-level radioactive waste is packaged according to waste form and disposed of in the intermediate-level radioactive waste vaults in E-Area. Long-lived radioactive waste is stored in the Long-Lived Waste Storage Building in E-Area until treatment and disposal technologies are developed (DOE 1998a).

Saltstone generated in the solidification of low-level radioactive waste salts extracted from high-level radioactive waste is disposed of in the Z-Area Saltstone Vaults. Saltstone is solidified grout formed by mixing the low-level radioactive waste salt with cement, fly ash, and furnace slag. Saltstone is the highest volume of solid low-level radioactive waste disposed of at SRS. SRS disposal facilities are projected to meet solid low-level radioactive waste disposal requirements, including low-level radioactive waste from off site, for the next 20 years (DOE 1996c).

3.3.11.5 Mixed Waste

The Federal Facilities Compliance Agreement addresses SRS compliance with RCRA Land Disposal Restrictions. The Agreement requires DOE facilities storing mixed radioactive waste to develop site-specific treatment plans and to submit them for approval (DOE 1996c). The site treatment plan for mixed radioactive waste specifies treatment technologies or technology development schedules for all SRS mixed radioactive waste (DOE 1998a). SRS is allowed to continue to generate and store mixed radioactive waste, subject to Land Disposal Restrictions. Schedules to provide compliance through treatment in the Consolidated Incinerator Facility are included in the Federal Facilities Compliance Agreement (DOE 1996c).

The SRS mixed radioactive waste program consists primarily of safely storing waste until treatment and disposal facilities are available. Mixed waste is stored in the A-, E-, M-, N-, and S-Areas in various tanks and buildings. These facilities include burial ground solvent tanks, the M-Area Process Waste Interim Treatment/Storage Facility, the Savannah River Technology Center Mixed Waste Storage Tanks, and the Defense Waste Processing Facility Organic Waste Storage Tank. These South Carolina Department of Health and Environmental Control-permitted facilities will remain in use until appropriate treatment and disposal is performed on the waste (DOE 1999h).

3.3.11.6 Hazardous Waste

Hazardous waste is accumulated at the generating facility for a maximum of 90 days, or stored in U.S. Department of Transportation-approved containers in three RCRA-permitted hazardous waste storage buildings and on three interim status storage pads in B- and N-Areas. Most of the waste is shipped off site to commercial RCRA-permitted treatment and disposal facilities using U.S. Department of Transportation-certified transporters. DOE plans to incinerate up to 9 percent of the hazardous waste (organic liquids, sludge, and debris) in the Consolidated Incinerator Facility (DOE 1996c). In 1995, 72 cubic meters (94 cubic yards) of hazardous waste were sent to onsite storage. Of this amount, 20 cubic meters (26 cubic yards) were shipped off site for commercial treatment or disposal (DOE 1999h).

3.3.11.7 Nonhazardous Waste

In 1994, the centralization and upgrading of the sanitary wastewater collection and treatment systems at SRS were completed. The program included the replacement of 14 (of 20) aging treatment facilities scattered across the site with a new 4,160 cubic meters (1.1 million gallons) per day central treatment facility, and connecting them with a new 29-kilometer (18-mile) sanitary sewer system. The central treatment facility treats sanitary wastewater by the extended aeration activated sludge process. The treatment facility separates the wastewater into two forms, clarified effluent and sludge. The liquid effluent is further treated by the nonchemical method of ultraviolet light disinfection to meet NPDES discharge limitations for the outfall to Fourmile Branch. The sludge is further treated to reduce pathogen levels to meet proposed land application criteria. The remaining sanitary wastewater treatment facilities are being upgraded as necessary by replacing existing chlorination treatment systems with nonchemical ultraviolet light disinfection systems to meet NPDES limitations (DOE 1996c).

SRS has privatized the collection, hauling, and disposal of its sanitary waste (DOE 1999h). SRS-generated solid sanitary waste is sent to the Three Rivers Landfill, a permitted disposal facility (DOE 2000). SRS disposes of other nonhazardous waste that consists of scrap metal, powerhouse ash, domestic sewage, scrap wood, construction debris, and used railroad ties in a variety of ways. Scrap metal is sold to salvage vendors for reclamation. Powerhouse ash and domestic sewage sludge are used for land reclamation. Scrap wood is burned on the site or chipped for mulch. Construction debris is used for erosion control. Railroad ties are shipped off site for disposal (DOE 1996c).

3.3.11.8 Waste Minimization

The total amount of waste generated and disposed of at SRS has been and continues to be reduced through the efforts of the pollution prevention and waste minimization program at the site. This program is designed to achieve continuous reduction of waste and pollutant releases to the maximum extent feasible and in accordance with regulatory requirements while fulfilling national security missions (DOE 1996c). The program focuses mainly on source reduction, recycling, and increasing employee participation in pollution prevention. For example, nonhazardous solid waste generation in 1995 was 32 percent below that of 1994, and the disposal volume of other solid waste, including radioactive and hazardous waste, was 38 percent below 1994 levels. In 1995, SRS achieved a 9 percent reduction in its radioactive waste generation volume compared with 1994. Total solid waste volumes have declined by more than 70 percent since 1991. Radioactive solid waste volumes have declined by about 63 percent, or more than 17,000 cubic meters (22,000 cubic yards), from 1991 through 1995. In 1995, more than 2,990 metric tons (3,300 tons) of nonradioactive materials were recycled at SRS, including 963 metric tons (1,062 tons) of paper and cardboard (DOE 1999h). The pollution prevention projects reduced the total amount of waste generated at SRS in 1997 by approximately 18,200 cubic meters (23,800 cubic yards) (DOE 1998b).

3.3.11.9 Preferred Waste Management Alternatives From the Final Waste Management Programmatic EIS and Associated Records of Decision

Preferred Alternatives from the Waste Management Programmatic EIS (DOE 1997a) are shown in **Table 3–26** for the four waste types analyzed in this EIS. Management of this waste could result in the construction of new waste management facilities at SRS and the closure of other facilities. Decisions on the various waste types were announced in a series of Records of Decision that have been issued on the Waste Management Programmatic EIS. The transuranic waste Record of Decision was issued on January 20, 1998 (63 FR 3629); the hazardous waste Record of Decision was issued on August 5, 1998 (63 FR 41810); the high-level radioactive waste Record of Decision on August 26, 1999 (64 FR 46661); and the low-level and mixed low-level radioactive waste Record of Decision on February 25, 2000 (61 FR 10061). The transuranic waste Record of Decision states, "...each of the Department's sites that currently has or will generate [sic]

transuranic waste will prepare and store its transuranic waste on site...." The hazardous waste Record of Decision states that most DOE sites will continue to use offsite facilities for the treatment and disposal of major portions of the nonwastewater hazardous waste, and Oak Ridge Reservation and SRS will continue to treat some of their own hazardous waste on site and in existing facilities where this is economically favorable. The high-level radioactive waste Record of Decision states that immobilized high-level radioactive waste will be stored at the site of generation. DOE decided in the Record of Decision for the management and disposal of low-level and mixed low-level radioactive waste to perform minimum treatment of low-level radioactive waste at all sites and continue, to the extent practicable, disposal of onsite low-level radioactive waste at INEEL, Los Alamos National Laboratory, the Oak Ridge Reservation, and SRS. For the management and disposal of mixed low-level radioactive waste, DOE decided to treat this waste at the Hanford site, INEEL, the Oak Ridge Reservation, and SRS, with disposal at the Hanford site and the Nevada Test Site. More detailed information concerning DOE's alternatives for the future configuration of waste management facilities at SRS is presented in the Waste Management Programmatic EIS and the hazardous, transuranic, high-level, and low-level and mixed low-level radioactive waste Records of Decision.

Table 3–26 Preferred SRS Waste Management Alternatives From the Waste Management Programmatic EIS and Associated Records of Decision

Waste Type	*Preferred Action*
High-level radioactive	DOE prefers onsite storage of SRS's immobilized high-level radioactive waste pending disposal in a geologic repository. [a]
Transuranic and mixed transuranic	DOE has decided that SRS should prepare and store its transuranic waste on site pending disposal at the Waste Isolation Pilot Plant. [b]
Low-level radioactive	DOE has decided to treat SRS low-level radioactive waste on site. SRS has been selected as one of the regional disposal sites for low-level radioactive waste. [c]
Mixed	DOE has decided to treat SRS mixed waste on site, including the possibility of treating mixed waste generated at other sites. SRS was not selected as one of the regional disposal sites for mixed waste. [c]
Hazardous	DOE has decided to use commercial and onsite SRS facilities for treatment of SRS nonwastewater hazardous waste, and to continue to use onsite facilities for wastewater hazardous waste. [d]

[a] From the Record of Decision for high-level radioactive waste (64 FR 46661).
[b] From the Record of Decision for transuranic waste (63 FR 3629).
[c] From the Record of Decision for low-level and mixed low-level radioactive waste (65 FR 10061).
[d] From the Record of Decision for hazardous waste (63 FR 41810).
Sources: DOE 1997a, 63 FR 3629, 63 FR 41810.

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4. ENVIRONMENTAL CONSEQUENCES

Chapter 4 describes the environmental consequences of the proposed action and alternatives to treat and manage sodium-bonded spent nuclear fuel. It begins with a general discussion of the expected environmental consequences; the product and waste forms that would be generated from the proposed action; and the methodology for assessing health effects from radiological and chemical effluent. It follows with a detailed description of the environmental consequences for the No Action and the reasonable alternatives. The chapter provides separate discussions on the environmental consequences of the intersite transportation of sodium-bonded spent nuclear fuel; the cumulative impacts at each of the proposed sites; and the programmatic considerations associated with the proposed action. The chapter concludes with a look at several issues under the proposed action, such as unavoidable, adverse environmental impacts; relationships between local, short-term uses of the environment and the enhancement of long-term productivity; and irretrievable commitments of resources.

4.1 OVERVIEW OF ENVIRONMENTAL CONSEQUENCES

This *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (SBSNF EIS) is in compliance with Council on Environmental Quality regulations that require the affected environment of proposed Federal actions to be "interpreted comprehensively to include the natural and physical environment and the relationship of people with the environment" (40 CFR 1508.14).

The environmental consequence analysis focused on potentially affected areas. These areas are discussed in detail: air quality, water resources, socioeconomics, public and occupational health and safety (normal operations and accident conditions), environmental justice, waste management, and transportation. For the remaining areas (i.e., land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources), analyses show that the proposed treatment activities would have minimal or no impact at the candidate sites regardless of the alternatives being considered. This is because existing facilities within developed areas would be used; no new land disturbance would take place and proposed activities would be consistent with current operations. Since none of the alternatives involve construction other than internal building modifications for installing new equipment, the effects of these modifications on any of the resources would be negligible and are not evaluated in this chapter.

The specific assumptions associated with the impact analysis common to all alternatives are provided in the appendices. The results of the assessment of environmental consequences are presented in this chapter. More detailed descriptions of the development of the impacts for some resource areas are presented in Appendices E through H, as follows:

- Appendix E, Evaluation of Human Health Effects From Normal Operations
- Appendix F, Evaluation of Human Health Effects From Facility Accidents
- Appendix G, Evaluation of Human Health Effects From Overland Transportation
- Appendix H, Environmental Justice Analysis

4.1.1 Presentation of the Environmental Consequences

The primary areas of concern are products and waste, impacts on the public, and occupational health and safety associated with the various sodium-bonded spent nuclear fuel treatment processes. Additional areas and topics covered in Chapter 4 include the following:

- Air Quality
- Water Resources
- Environmental Justice
- Socioeconomics
- Waste Management
- Transportation Impacts
- Short-term versus Long-term Resource Commitments
- Irreversible and Irretrievable Resource Commitments
- Cumulative Impacts

Several kinds of impacts are not discussed in Chapter 4 because they would not occur, they would be extremely small, and/or they are covered by other analyses:

Land Use—The treatment and management of sodium-bonded spent nuclear fuel would not require the construction of new facilities on previously undisturbed land at Argonne National Laboratory-West (ANL-W) or the Savannah River Site (SRS).

Intrasite Transportation—The incident-free impacts of intrasite transportation would be limited to radiation exposure to workers loading and unloading trucks and are included in the overall worker dose values presented for each process. The accident risks are bounded by the site accident risk analysis. Strict site safety procedures and short travel distances would limit the impacts to workers.

Noise—Noise impacts at the management sites would be minor and would be limited to noise generated during operations. No offsite noise impacts are expected except for minor changes in traffic noise levels.

Ecological Resources—Because no new construction in undisturbed areas would be required for the treatment and management of sodium-bonded fuel, there would be no disturbance to terrestrial and aquatic habitats or wetlands. Thus, there would be no negative impacts from construction on terrestrial or aquatic plants or animals, including threatened and endangered species.

Scientific evidence indicates that limiting chronic radiation doses to below 0.1 rad per day to the more radiosensitive species in terrestrial ecosystems provides adequate protection for the population. In the aquatic environment, limiting the chronic dose rate to 1 rad or less per day to an individual member of the aquatic population would provide adequate protection for the population (IAEA 1992). Therefore, limiting chronic radiation doses to below 0.1 rad per day would not harm animal or plant populations. This is equivalent to a dose of 100 millirem per day from direct radiation (inhalation dose) to an individual. Compliance with U.S. Department of Energy (DOE) Order 5400.5 to limit the exposure of the most exposed member of the public (a hypothetical individual residing at the site boundary) to 100 millirem per year (i.e., about 0.3 millirem per day from all pathways) and to 10 millirem per year from the air pathway (40 CFR 61 dose limit) makes it highly probable that dose rates to plants and animals in the same area would be less than 0.1 rad per day. The maximum annual dose to the most exposed member of the public under any one of the alternatives analyzed would be a small fraction (about 0.2 percent) of 1 millirem. Therefore, no radiological damage to plant and animal populations would be expected to result from the sodium-bonded spent nuclear fuel treatment processes.

Chemicals emitted to the environment during routine processing activities are presented under each alternative. These releases are essentially independent of the process being performed. They are generated from operation of support facilities, such as operation of emergency diesel generators during testing and/or fuel burning for facility heating and power production. The quantities of releases attributable to the treatment of the sodium-bonded spent nuclear fuel would be very small fractions of the current releases from each management site.

The site environmental reports did not identify any measurable impacts on plants or animals because the amounts emitted are very low or the chemicals have little potential for causing negative effects.

For the reasons discussed above, no adverse impacts to ecological resources would be expected to occur due to DOE's treatment and management of sodium-bonded spent nuclear fuel.

Cultural and Paleontological Resources—No new facilities would be needed or constructed, therefore, there would be no impacts on cultural or paleontological resources.

Geology and Soils—No new facilities would be needed or constructed. Therefore, there would be no disturbance to either geologic or soil resources at the management sites. Hazards from large-scale geologic conditions were analyzed in detail in various DOE programmatic environmental impact statements (EISs) and site-specific facility safety analysis reports. The impacts from these hazards (e.g., earthquakes) on the management facilities and treatment processes are evaluated in this EIS.

4.1.2 Products and Waste

Generation and Disposition—All of the treatment processing alternatives presented in this EIS, except for direct disposal in high-integrity cans, would change the sodium-bonded spent nuclear fuel into other forms. Driver and blanket sodium-bonded spent nuclear fuel would be input—products and waste would be the output. The products and waste would be better suited for storage, transportation, and disposal or other disposition than the existing sodium-bonded fuel. The products and waste fall into several distinct categories:

- Materials to be managed as high-level radioactive waste would be generated at SRS and/or ANL-W. The final form would be ceramic, metallic, a melt and dilute product, or borosilicate glass inside stainless steel canisters. The production of ceramic, metallic, and melt and dilute products at ANL-W would result mainly from the transformation of spent nuclear fuel to a different form that would make the final product more stable and lead to an overall reduction in repository volume need. This waste would be stored at SRS and/or ANL-W until a geologic repository is ready to receive it.

- Transuranic waste refers to processed materials that contain alpha-emitting material (such as plutonium) with radioactivity concentrations above 100 nanocuries per gram of waste. Transuranic waste would be generated from all treatment technologies. This waste could be disposed of in the Waste Isolation Pilot Plant.

- The separated uranium resulting from the electrometallurgical treatment process at ANL-W would be made into solid metal ingots. The separated uranium resulting from processing the driver spent nuclear fuel would be made into low-enriched uranium ingots. The ingots would be more than 99.7 percent pure uranium; the balance of the material would be mainly zirconium (the alloy in the fuel) and trace quantities of fission products and actinides requiring additional purification before the uranium ingots could be used commercially. The uranium ingots would be stored in secure facilities along with other uranium already in storage at ANL-W until decisions are made about their disposition.

- Separated depleted uranium from plutonium-uranium extraction (PUREX) processing of declad and cleaned blanket spent nuclear fuel at SRS would be made into uranium oxides and stored in drums along with other depleted uranium at SRS (more than 27,000 metric tons of depleted uranium are stored currently at SRS). The 57 metric tons of depleted uranium that would be separated in this process would be a small fraction of what is stored currently.

- Separated plutonium resulting from PUREX processing of declad and cleaned blanket spent nuclear fuel at F-Canyon would be in a metallic form. The separated plutonium, less than 260 kilograms (572 pounds),

would be stored in secure facilities along with the plutonium already in storage (about 2.4 metric tons) at SRS. The plutonium would be disposed of in accordance with the Record of Decision (75 *FR* 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999d).

- Low-level radioactive waste would be generated from all treatment technologies considered. This waste would be disposed of in existing facilities using routine procedures.

- Saltstone would be generated only at SRS. Saltstone is a form of concrete containing low levels of radioactivity and would be disposed of on site.

- Mixed waste would be generated at ANL-W and SRS. At ANL-W, mixed waste would be generated mainly from cadmium contamination, which would be present in one of the electrorefiners. It would consist of waste categorized as indirect process solid waste and would include discarded equipment and materials from decontamination operations. At SRS, liquid mixed waste would be generated from contamination by various chemicals in the dissolution and extraction facilities.

Waste Minimization—DOE would incorporate the best available practices into all the processing technologies at the two management sites to generate the smallest possible amount of waste. The DOE sites managing the sodium-bonded spent nuclear fuel would comply with DOE's waste minimization and pollution prevention goals. The following summarizes recent achievements in pollution prevention and waste minimization at ANL-W and SRS:

- ANL-W conducted pollution prevention projects in 1997 that reduced waste generation by an estimated 1,700 cubic meters (61,100 cubic feet) at a cost savings of $154,000. Radioactive waste generation in 1997 was reduced by 61 percent compared to 1993 baseline levels. Mixed waste generation was increased by 67 percent, hazardous waste generation was reduced by 44 percent, and sanitary waste generation was reduced by 32 percent compared to baseline levels. Fifty-six percent of sanitary waste was recycled in 1997. ANL-W affirmative procurement purchases are not tracked separately, and are included in the Idaho National Engineering and Environmental Laboratory (INEEL) totals. For INEEL, 72 percent of the materials purchased were U.S. Environmental Protection Agency (EPA)-designated recycled products (DOE 1998e).

- SRS conducted pollution prevention projects in 1997 that reduced waste generation by an estimated 18,200 cubic meters (644,000 cubic feet) at a cost savings of $18.5 million. Radioactive waste generation in 1997 was reduced by 57 percent compared to 1993 baseline levels. Mixed waste generation was increased by 115 percent, hazardous waste generation was reduced by 15 percent, and sanitary waste generation was reduced by 58 percent compared to baseline levels. Seventy-eight percent of sanitary waste was recycled in 1997, and 52 percent of the materials purchased under the affirmative procurement process were EPA-designated recycled products (DOE 1998e).

4.1.3 Radiological and Chemical Health Risk Estimates

The methodologies used to evaluate potential radiological and chemical health effects from operational effluent are described in Appendix E. This section provides information about the development and interpretation of the health risk estimates.

Radiological—The effect of radiation on people depends upon the kind of radiation exposure (alpha, beta, and neutron particles and gamma and x-rays), duration of exposure, and the total amount of tissue exposed to radiation. The amount of radiant energy imparted to tissue from exposure to ionizing radiation is referred to as "absorbed dose." The sum of the absorbed dose to each tissue, when multiplied by certain quality and

weighting factors that take into account radiation quality and different sensitivities of these various tissues, is referred to as "effective dose equivalent."

An individual may be exposed to radiation from outside or inside the body, because radioactive materials may enter the body by ingestion or inhalation. External dose is different from internal dose in that it is delivered only during the actual time of exposure. An internal dose, however, continues to be delivered as long as the radioactive source is in the body (although both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time). The dose from internal exposure is calculated over 50 years following the initial exposure.

The regulatory annual radiation dose limits to the maximally exposed offsite individual from total operations at a DOE site are 10 millirem from atmospheric pathways, 4 millirem from drinking water pathways, and 100 millirem from all pathways combined (DOE Order 5400.5 and 40 CFR Part 61, Subpart H). The potential doses associated with the normal operation of various treatment technologies and storage of sodium-bonded spent nuclear fuel would be very small fractions of these values, and total site doses would remain well within these DOE limits. For comparison, DOE estimates that the average individual in the United States receives a dose of approximately 360 millirem per year from all radiation sources combined, including natural and medical sources (see Appendix E, Section E.2.1, for details).

The collective or "population" dose to an exposed population is calculated by summing the estimated doses received by each member of the exposed population. The collective dose received by the exposed population is measured in person-rem. For example, if 1,000 people each received a dose of 0.001 rem, the population dose would be 1 person-rem (1,000 persons × 0.001 rem = 1 person-rem). The same population dose (1 person-rem) would result if 500 people each received a dose of 0.002 rem (500 persons × 0.002 rem = 1 person-rem).

Radiation can cause a variety of adverse health effects in people. A large dose of radiation can cause prompt death. At low doses of radiation, the most important adverse health effect from environmental and occupational radiation exposures (which are typically low doses) is the potential inducement of fatal cancers. This effect is referred to as "latent" cancer fatalities because the cancer may take years to develop and for death to occur.

In addition to latent cancer fatalities, other health effects could result from exposures to radiation. These effects include nonfatal cancers among the exposed population and genetic effects in subsequent generations. The dose-to-effect factors for fatal and nonfatal cancers are shown in **Table 4–1**. As indicated in this table, the nonfatal cancers and genetic effects are less probable consequences per unit of radiation exposure. For simplicity, this EIS presents estimated effects of radiation only in terms of latent cancer fatalities. Estimates of the total detriment (fatal cancers, nonfatal cancers, and genetic effects) due to radiation exposure may be obtained from the estimates of latent cancer fatalities presented in this EIS by multiplying by 1.4 for workers and by 1.46 for the general public.

The factors used in this EIS to relate a dose to its effect are 0.0004 lifetime probability of a latent cancer fatality per person-rem for workers and 0.0005 lifetime probability of a latent cancer fatality per person-rem for individuals among the general population. The latter factor is slightly higher because some individuals in the public, such as infants and children, are more sensitive to radiation than workers. These factors are based on the 1990 Recommendations of the International Commission on Radiological Protection (ICRP 1991) and are consistent with those used by the U.S. Nuclear Regulatory Commission (NRC) in its rulemaking *Standards for Protection Against Radiation* (10 CFR 20). The factors apply where the dose to an individual is less than 20 rem and the dose rate is less than 10 rem per hour. At higher doses and dose rates, the factors used to relate radiation doses to latent cancer fatalities are doubled. At much higher doses, prompt effects, rather than latent cancer fatalities risk, may be the primary concern.

Table 4–1 Risk of Latent Cancer Fatalities and Other Health Effects From Exposure to 1 Rem of Radiation[a]

Individual [b]	*Latent Cancer Fatalities*	*Nonfatal Cancers*	*Genetic Effects*	*Total Detriment*
Worker	0.0004	0.00008	0.00008	0.00056
Public	0.0005	0.0001	0.00013	0.00073

[a] When applied to an individual, units are lifetime probability of a latent cancer fatality per rem of radiation dose. When applied to a population of individuals, units are the excess number of cancers per person-rem of radiation dose. Genetic effects as used here apply to populations, not individuals.

[b] The difference between the worker risk and the general public risk is attributable to the fact that the general population includes more individuals in the more sensitive age group of less than 18 years of age.

Note: One rem equals 1,000 millirem.

Sources: NCRP 1993, ICRP 1991.

These factors are used to calculate the statistical expectations of the effects of exposing a population to radiation. For example, if 100,000 people each were exposed to a one-time radiation dose of 100 millirem (0.1 rem), the collective dose would be 10,000 person-rem. The exposed population then would be expected to experience 5 additional latent cancer fatalities from the radiation (10,000 person-rem × 0.0005 lifetime probability of a latent cancer fatality per person-rem = 5 latent cancer fatalities).

Sometimes calculations of the number of latent cancer fatalities associated with radiation exposure do not yield whole numbers and, especially in environmental applications, may yield numbers less than 1. For example, if 100,000 people each were exposed to a total dose of only 1 millirem (0.001 rem), the collective dose would be 100 person-rem, and the corresponding estimated number of excess latent cancer fatalities would be 0.05 (100,000 persons × 0.001 rem × 0.0005 latent cancer fatalities per person-rem = 0.05 latent cancer fatalities). The "0.05" means that there is one chance in 20 that the exposed population would experience one latent fatal cancer. In other words, the latent cancer fatality rate of 0.05 is the *expected* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people. In most groups, nobody (0 people) would incur a latent cancer fatality from the 1 millirem dose each member would have received. In a small fraction of the groups, one latent cancer fatality would result; in exceptionally few groups, two or more latent cancer fatalities would occur. The average expected number of deaths for all the groups would be 0.05 latent cancer fatalities (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is 0 latent cancer fatalities.

The same concept is applied to estimate the effects of continuous radiation exposure to an individual member of the public. Consider the effects of an individual's exposure to a 360-millirem (0.36-rem) annual dose from all radiation sources (natural and medical). The probability that the individual would develop a latent fatal cancer from continuous exposure to this radiation over an average life of 72 years (presumed) is 0.013 (1 person × 0.36 rem per year × 72 years × 0.0005 latent cancer fatality risk per person rem = 0.013), or one chance in 77 that the individual would develop a fatal cancer from this radiation exposure.

The estimates of health effects from radiation doses used in this EIS are based on the linear no-threshold theory of radiation carcinogenesis, which postulates that all radiation doses, even those close to 0, are harmful. As explained in Appendix E, the numerical estimates of fatal cancers presented in this EIS were obtained from the nominal risk estimated for lifetime total cancer mortality, resulting from a dose of 0.1 gray (10 rad) (National Research Council 1990). Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of fatal cancers. Studies of human populations exposed to low doses are inadequate to demonstrate the actual level of risk. There is a scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation, and the possibility of no risk cannot be excluded (CIRRPC 1992).

This EIS provides radiation dose estimates and probabilities of latent cancer fatalities (risks) for various receptors from management facility radiation exposure during normal operations and accident conditions. The receptors are defined as follows:

Worker – An individual actively participating and/or supporting the operation of the facility.

Noninvolved worker– An individual who is not involved in the operation of the facility. For estimating the impact, the individual is assumed to be 100 or more meters (330 or more feet) from the radioactive or chemical material release point.

Maximally exposed offsite individual – An individual member of the public assumed to be residing at the site boundary who could receive the maximum dose from radiation or hazardous chemicals.

Population – members of the general public residing within an 80-kilometer (50-mile) radius of the facility.

For incident-free (normal) operations, the EIS provides two sets of impacts (dose and risk): maximum annual and project total impacts for all alternatives. The maximum annual impacts result from simultaneous treatment of both driver and blanket sodium-bonded spent nuclear fuel in a given year, and the project total impacts represent the overall impacts from treatment of all sodium-bonded spent nuclear fuel. For accident conditions, the EIS provides both the consequence (dose) per accident and the associated risk.

Chemical—The potential impacts of exposure to hazardous chemicals released to the atmosphere as a result of the processing of sodium-bonded spent nuclear fuel were evaluated for incident-free operations and accident conditions at management facilities. Small amounts of hazardous and toxic chemical releases would be expected from incident-free operation of the treatment technology support facilities and equipment (e.g., auxiliary steam power house, diesel generators). The health effects from these releases were calculated for the maximally exposed offsite individual (an individual member of the public residing at the site boundary). The health effects evaluated in this analysis include excess latent cancer fatalities and chemical-specific noncancer health effects. The maximally exposed offsite individual was assumed to be located in the region with the highest estimated concentration. The health effects from releases of hazardous chemicals during accident conditions were evaluated in terms of comparison to Emergency Response Planning Guideline (ERPG) values. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate to observe adverse effects (see Appendix F, Section F.3.1.2, for more detail).

4.2 NO ACTION ALTERNATIVE

Under the No Action Alternative, the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed from the fuel). Under this alternative, two options were considered.

The EIS evaluates:

a. The impacts from the activities required to monitor and stabilize the sodium-bonded spent nuclear fuel as necessary for continued safe and secure storage at current locations, or until a new treatment technology, such as the glass material oxidation and dissolution system (GMODS) or plasma arc, is developed (see Section 2.6 for more details on GMODS and plasma arc technology development needs).

b. The impacts from direct disposal of sodium-bonded spent nuclear fuel in a geologic repository by packaging the fuel in high-integrity cans without sodium removal. At the present time, direct disposal of sodium-bonded spent nuclear fuel is precluded by DOE policy concerning acceptance of Resource Conservation and Recovery Act (RCRA)-designated mixed waste (which contains both hazardous and radioactive waste).

Under either option of the No Action Alternative, the EIS evaluates the impacts associated with activities required to clean and stabilize the waste materials generated during the Electrometallurgical Treatment Research and Demonstration Project at ANL-W. As part of this demonstration project, approximately 1.6 metric tons of heavy metal of Experimental Breeder Reactor -II (EBR-II) fuel consisting of about 1.2 metric tons of blanket spent nuclear fuel and 0.4 metric tons of driver spent nuclear fuel were processed (DOE 1996b). The waste materials generated in this project currently are being transformed into ceramic and metallic waste forms. This process will continue until all of this waste is transferred to ceramic and metallic waste forms. The remaining sodium-bonded spent nuclear fuel in the treatment facilities will be packaged and transferred to dry storage in the Radioactive Scrap and Waste Facility.

DOE also is transferring to dry storage all INEEL spent nuclear fuel, including the sodium-bonded spent nuclear fuel currently stored at Idaho Nuclear Technology and Engineering Center (INTEC) Building 603 (wet storage basin). During this transfer, each fuel can containing sodium-bonded fuel will be nondestructively examined to determine the fuel can condition and its suitability for storage. If any fuel can is found to be degraded, resulting in water in-leakage, it will be repackaged and transferred to ANL-W for stabilization and/or repackaging for storage. The fuel transfer activities are planned for completion by December 2000. The sodium-bonded spent nuclear fuel currently stored at INTEC Building 666 (wet storage basin) will remain in the basin until the planned defueling and facility closure in the year 2023. These fuel movement activities would be performed independently of the activities within this EIS.

About 5 metric tons of heavy metal of EBR-II blanket spent nuclear fuel contained in 107 storage cans currently stored at the Radioactive Scrap and Waste Facility do not meet the long-term confinement requirements. Under the No Action Alternative continued safe storage option. These storage cans would be brought to the Hot Fuel Examination Facility to be repackaged in more durable storage liners and would be returned to storage. These activities, along with the waste processing activities at ANL-W, would be completed in about two years after the necessary waste handling equipment is installed. The sodium-bonded spent nuclear fuel that currently is stored at INTEC would remain there, and sodium-bonded spent nuclear fuel transferred to Idaho in the future, as specified in the amended Record of Decision for the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (Programmatic Spent Nuclear Fuel EIS) (61 FR 9441), also would be stored at INTEC. Consistent with the DOE-State of Idaho Settlement Agreement and Consent Order, all spent nuclear fuel would need to be transferred out of the State of Idaho by January 1, 2035. Under this option of the No Action Alternative, in the event that sodium-bonded spent nuclear fuel has not been treated before 2035, DOE would package the stored fuel at ANL-W and transfer it to the INEEL Dry Transfer facility. DOE also may decide to use the facilities at ANL-W to package the sodium-bonded spent nuclear fuel stored at INTEC. In the event that the sodium-bonded spent nuclear fuel has not been treated before 2035, the stored fuel would be removed from the State of Idaho by the year 2035. The environmental impacts of untreated sodium-bonded spent nuclear fuel removal would be evaluated in a separate National Environmental Policy Act (NEPA) document.

Under the No Action Alternative direct disposal option, all sodium-bonded spent nuclear fuel at INTEC would be transferred to ANL-W and repackaged in high-integrity cans in preparation for direct disposal. The activities associated with the preparation of sodium-bonded spent nuclear fuel for direct disposal would be similar to those needed to prepare the fuel for continued safe storage. The activities for direct disposal would occur sometime after those for the continued storage option. This is because a decision to directly dispose of the

sodium-bonded spent nuclear fuel in a geologic repository would be made only after it was determined that it would meet the repository acceptance criteria. Currently, there are no acceptance criteria for this fuel type. If direct disposal of the sodium-bonded spent nuclear fuel becomes possible, DOE would use the facilities at ANL-W to prepare all sodium-bonded spent nuclear fuel at the INEEL site. Preparation of driver spent nuclear fuel for direct disposal requires consideration of criticality safety, thereby limiting the amount of driver spent nuclear fuel that could be packaged in a canister. This would lead to larger repository volume needs per unit mass for driver fuel.

The activities in this option would include:

1. Repackaging 107 cans containing 5 metric tons of heavy metal of blanket spent nuclear fuel in the first two years (ending in 2003); see the continued storage option above.

2. Transferring sodium-bonded spent nuclear fuel currently stored at INTEC (Building 666 Basin and Building 603 Dry Storage) to ANL-W between 2003 and 2023. The 2023 date corresponds to the target date for closure of Building 666 Basin at INTEC. Under this assumption, the fuel in Building 666 Basin would be in wet storage for 23 years.

3. Repackaging the spent nuclear fuel at ANL-W in high-integrity cans to meet the target date for fuel transfer out of the State of Idaho (January 1, 2035). All sodium-bonded spent nuclear fuel would be transferred to the Hot Fuel Examination Facility for characterization and placement in high-integrity cans. The preparation and canning activities would be completed in about three years. The canned fuel would be stored temporarily at the Radioactive Scrap and Waste Facility. The stored fuel cans would be packaged in standardized canisters and transferred to the INEEL Dry Transfer Facility for packaging and shipment to the repository.

The environmental impacts for both options under the No Action Alternative are presented below. Where the impacts are different between the options, two sets of results are presented.

4.2.1 Air Quality

Nonradiological Gaseous Emissions

As explained in Appendix E, Section E.5.3.1, under the proposed action and either option of the No Action Alternative, small quantities of criteria pollutants and hazardous chemicals are generated from the operation of the emergency diesel generators supporting both the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W. The emissions from these generators are independent of any of the treatment processes under the proposed action and the No Action Alternative addressed in this EIS.

Table 4–2 summarizes the concentrations of criteria and hazardous air pollutants. The concentrations are compared to their corresponding ambient air quality standards. Only those air pollutants that are expected and have ambient air quality standards are presented in the table. The emissions are generated from diesel generators currently in operation and are considered as part of the baseline concentration. No increases in emissions are expected under the No Action Alternative. Therefore, a Prevention of Significant Deterioration increment analysis was not required. In addition, the INEEL site is located in areas of attainment for the criteria pollutants; therefore, no conformity analysis is required.

Table 4–2 Nonradiological Air Quality Concentrations at the Site Boundary Under the No Action Alternative for Comparison With Ambient Air Quality Standards

	Averaging Period	*Most Stringent Standard or Guideline (micrograms per cubic meter)* [a]	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutant			
Carbon monoxide	8 hours 1 hour	10,000 40,000	32.7 46.8
Nitrogen dioxide	Annual	100	2.8
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 150 150	0.01 0.19 Not available
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.45 11.50 25.80
Hazardous and Toxic Compounds			
1,3-Butadiene	Annual	0.0036	0.0000355
Acetaldehyde	Annual	0.45	0.0000226
Acrolein	24 hours	12.5	0.000181
Benzene	Annual	0.12	0.000694
Formaldehyde	Annual	0.077	0.0000709
Toluene	24 hours	18,750	0.00664
Xylene	24 hours	21,750	0.00447

PM_n = Particulate matter less than or equal to *n* microns in diameter.

[a] The standards for hazardous and toxic compounds apply only to increases in emissions from new or modified sources and are provided for information purposes only, as the concentrations from releases at ANL-W under all alternatives are not expected to increase.

Radiological Gaseous Emissions

Potential radiological releases from sodium-bonded spent nuclear fuel would be very small under both options of this alternative. Under both options, the spent nuclear fuel would remain stored in sealed canisters while at INEEL (i.e., INTEC or ANL-W) until 2035. However, degradation of sodium-bonded spent nuclear fuel or its enclosure (e.g., a sealed canister) during storage cannot be ruled out. It is expected that a small fraction of the fuel would degrade during storage, allowing its gaseous fission products to enter the storage canister. These fission gases would be released to the environment only if the sealed canister were to fail or be opened during fuel handling for examination and repackaging. As detailed in Appendix E, Section E.4.6, current experience at INTEC and ANL-W indicates very small fuel degradation problems during the storage period. It was estimated that, over 30 years of storage, about 1 percent of the fuel would be in degraded condition while in dry storage and about 3 percent of the fuel would fail while in wet storage. While in dry storage, there would be no releases of gaseous fission products to the environment. The fission gases would be released to the environment only during fuel repackaging. In wet storage, fuel canister degradation and resulting fuel failure would lead to releases of gaseous fission products. The estimated gaseous fission product releases during the entire period (over 35 years) of the No Action Alternative would be 51 curies of tritium oxides, 760 curies of krypton-85, and 0.000018 curies of iodine-129.

4.2.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Discharge waters to the Industrial Waste Pond or to the Sanitary Sewage Lagoons are not waters of the United States and are exempt from compliance under the National Pollutant Discharge Elimination System (NPDES). However, these are designated as waters of the State of Idaho and, as such, require compliance with State regulations that govern application of nonhazardous liquid waste (i.e., Land Application Permits). ANL-W has applied to the State of Idaho for Land Application Permits for the Industrial Waste Pond and Ditches and the Sanitary Waste Treatment Pond Land Application Area (DOE 1996a, DOE 1998c). ANL-W routinely monitors the effluent discharges to make sure they are within those limits identified in the Land Application Permits. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W.

Radiological Liquid Effluent

No radiological liquid effluent would be discharged to the surface water.

Groundwater

Under either option of this alternative, there would be some reduction in groundwater consumption for domestic uses, since the number of workers at ANL-W is expected to decrease. The current water use at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

For either option of this alternative, no nonradiological liquid effluent or waste would be discharged to groundwater.

Radiological Liquid Effluent

For either option of this alternative, no radiological liquid effluent would be discharged to groundwater.

4.2.3 Socioeconomics

Under either option of the No Action Alternative, there could be a reduction of approximately 350 workers at ANL-W if a treatment technology is not selected or the decision is delayed. If all of these workers were to leave the regional economic area, this could result in the loss of an additional 940 indirect jobs in the economic region. The total potential loss of about 1,290 jobs represents less than a 1 percent decrease in civilian employment in the regional economic area, which was estimated to be 150,403 in 1996 (DOE 1999d).

Since any reduction in the ANL-W labor force under the No Action Alternative would take place over time, combined with the fact that many of these workers could also support other missions at INEEL, the effects are expected to be gradual. By 2010, the contributory effect of this and the potential for beneficial effects

from other industrial and economic sectors within the regional economic area would serve to reduce or mask any effect on the regional economy. Neither option of the No Action Alternative, therefore, would result in any noticeable change in the existing regional economy, population and housing characteristics, or community services within the region of influence at ANL-W (see Section 3.2.8).

4.2.4 Public and Occupational Health and Safety

The assessments of potential radiological and chemical impacts associated with the No Action Alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–3 and 4–4 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–6 and 4–7. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–8. Background information on the effects of radiation on human health and safety is presented in Section 4.1.3 and Appendix E, Section E.2.

4.2.4.1 Normal Operations

Radiological Impacts

Under either option of the No Action Alternative, radioactive releases from normal operations associated with spent nuclear fuel storage activities at ANL-W and INTEC would be small. The releases would occur from fuel degradation in wet storage and during fuel handling. Under both options, the same amount of gaseous radioactive material would be released. As explained in Appendix E, Section E.4.6, under both options, some fuel would be repackaged at the beginning in the first two years and all of the fuel would be repackaged by 2035 prior to shipment outside the INEEL site. The repackaging would occur over a three-year period. Releases would occur both from INTEC during wet storage and from ANL-W during fuel handling and repackaging operations. However, since INTEC is further away from the INEEL site boundary and major population centers compared to ANL-W, the releases were assumed to occur from ANL-W, thereby maximizing the impacts.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–3**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public would be much smaller than the limit of 10 millirem per year limit set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

The average worker dose (for ANL-W and INTEC workers) under the No Action Alternative was estimated to be similar to that currently experienced at ANL-W. Under both options, waste and fuel handling and repackaging activities would occur over a 5-year period, with standby operations for the remaining 30 years. One additional year also would be necessary to deactivate the facility. During fuel handling operations, the estimated annual total worker population dose would be 22 person-rem; during storage (standby) operations, it would be 2.2 person-rem; and during deactivation, it would be 33 person-rem, for a total of 209 person-rem over 35 years (see **Table 4–4**).

Table 4–3 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under the No Action Alternative

Receptor	*Impacts*
Population Within 80 Kilometers (50 Miles) in the Year 2010	
Collective dose (person-rem per year) [a]	0.0015
Excess latent cancer fatalities (per year)	7.5×10^{-7}
Project total excess latent cancer fatalities [b]	6.5×10^{-6}
Maximally Exposed Offsite Individual	
Dose (millirem per year) [a]	0.00026
Percent of annual background [c]	0.000072
Latent cancer fatality risk (per year)	1.3×10^{-10}
Project total lifetime cancer fatality risk [b]	1.1×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)	
Dose (millirem per year) [d]	6.2×10^{-6}
Latent cancer fatality risk	3.1×10^{-12}
Project total lifetime cancer fatality risk [b]	2.7×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over 35 years.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–4 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under the No Action Alternative

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over 35 years)	0.00084
Worker Population	
Collective dose (person-rem per year)	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	209
Project total excess latent cancer fatalities	0.084

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

Source: ANL 1999.

As shown in Tables 4–3 and 4–4:

- The maximum annual dose to the maximally exposed offsite individual would be 0.00026 millirem, with an associated risk of developing a lifetime fatal cancer of 1.3×10^{-10} per year (or one chance in 7.7 billion that the individual would develop a fatal cancer per year of exposure).

- The collective maximum annual dose to the population within 80 kilometers (50 miles) of the storage facilities at ANL-W would be 0.0015 person-rem, with an associated 7.5×10^{-7} latent cancer fatalities per year (or one chance in 1.3 million that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of ANL-W would be 6.5×10^{-6} latent cancer fatalities (or one chance in 154,000 that the exposed population would experience a fatal cancer).

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 209 person-rem with an associated 0.084 latent fatal cancers (or one chance in 12 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alterative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under either option of this alternative are summarized in **Table 4–5**. Appendix E, Section E.5, provides details on the model used and results obtained. The results (presented in Table 4–5) indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

Table 4–5 Hazardous Chemical Impacts to the Public From Operational Activities Under the No Action Alternative

Chemical	*Maximum Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
1,3-Butadiene	3.6×10^{-8}	None	9.9×10^{-9}
Acetaldehyde	2.3×10^{-8}	2.5×10^{-6}	5.0×10^{-11}
Acrolein	7.1×10^{-9}	0.00035	None
Benzene	6.9×10^{-7}	None	5.4×10^{-9}
Formaldehyde	7.1×10^{-8}	None	9.2×10^{-10}
Toluene	2.5×10^{-7}	6.2×10^{-7}	None
Hazard Index		0.00036	Not applicable

4.2.4.2 Facility Accidents

The potential radiological impacts to the public and noninvolved onsite workers due to accidents are summarized in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, an earthquake, and an aircraft crash.

Under either option of the No Action Alternative, spent nuclear fuel transfer and waste processing activities associated with cleaning and stabilizing the waste materials generated during the Electrometallurgic

Treatment Research and Demonstration Project at ANL-W would be performed. These activities would have the potential to involve accident scenarios similar to those evaluated for Alternative 1 as presented in Section 4.3.4.2. However, the consequences associated with these accident scenarios would be lower because of the limited quantities of waste to be stabilized. Accidents associated with spent nuclear fuel transfer activities also could occur during fuel removal from the Radioactive Scrap and Waste Facility and packaging for offsite shipment to a repository. These accidents would lead to consequences similar to those evaluated for Alternative 1 as presented in Section 4.3.4.2. It is estimated that the spent nuclear fuel transfer and waste stabilization activities would occur over a two-year period. Fuel handling and repackaging for offsite shipment would occur over a three-year period.

No reasonably foreseeable accident scenarios could be identified that would impact sodium-bonded spent nuclear fuel in dry storage at the Radioactive Scrap and Waste Facility or in wet or dry storage at INTEC. In storage, the sodium-bonded spent nuclear fuel is in a safe and stable configuration. Generally, the only activity associated with the stored spent nuclear fuel is monitoring of the fuel and the storage facility. While in storage, activities that could lead to accidents (movement, repackaging, or processing of the spent nuclear fuel) are not performed. However, approximately 1.2 metric tons of sodium-bonded spent nuclear fuel currently in wet storage in Basin 603 at INTEC would be transferred to dry storage facilities at INTEC. Handling accidents could occur during transfer activities at INTEC similar to the accident scenarios evaluated for ANL-W. Because INTEC is further away from the INEEL site boundary and major population centers compared to ANL-W, the health impacts from accidents at INTEC would be less than those from similar accidents at ANL-W.

Table 4–6 presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker. The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. The 50th percentile condition represents the median meteorological condition, and is defined as that for which more severe conditions occur 50 percent of the time. The 95th percentile condition represents relatively low probability meteorological conditions that produce higher calculated exposures, and is defined as that condition that is not exceeded more than 5 percent of the time. DOE did not quantitatively estimate the involved worker dose due to accidents. The consequences to involved workers are qualitatively assessed. This approach is used for the following two reasons: (1) no adequate method exists for calculating meaningful consequences at or near the location where the accident occurs, and (2) safety assurance for facility workers is demonstrated by both the workers' training and by the establishment of an Occupational Safety and Health Administration (OSHA) process safety management system (29 CFR 1910.119). In any accident scenario, the individuals most likely to be injured are the involved workers. The risk to these workers would be due to both radiological and nonradiological effects. In a fire, the involved workers could be exposed to airborne radioactive material, in addition to the smoke and heat of the fire. In an explosion, there could be flying debris and containment barriers could be broken, exposing workers to airborne radioactive material. Most spills would not have a major effect on involved workers because they would clean up the spill wearing protective clothing and respirators as necessary. An accidental criticality could expose involved workers to large doses of prompt penetrating radiation, which could cause death in a short period of time. An earthquake accident could present a very severe nonradiological effect to the involved workers. For example, in a beyond-design-basis earthquake, the workers are likely to be hurt or could be killed from the collapse of the building before they could be evacuated (see Appendix F, Section F.2.2.2, for more detail). The accident risks are summarized in **Table 4–7**.

Table 4–6 Accident Frequency and Consequences Under the No Action Alternative

Accident [a]	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [b]	Dose (person-rem)	Excess Latent Cancer Fatalities [c]	Dose (millirem)	Latent Cancer Fatality Risk [b]
Salt powder spill in the Hot Fuel Examination Facility cell [d]	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Cask drop during spent nuclear fuel transfer	0.01	0.03	1.5×10^{-8}	0.0035	1.7×10^{-6}	0.00084	3.4×10^{-10}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.00070	4.7	1.9×10^{-6}
Salt transfer drop	1×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Beyond-design-basis earthquake	0.00001	96	0.000048	11	0.0055	37	0.000015

[a] Only accidents involving EBR-II driver spent nuclear fuel, which maximizes the consequences, are presented.
[b] Increased likelihood of a latent cancer fatality.
[c] Increased number of latent cancer fatalities.
[d] The salt powder spill was assumed to have similar characteristics to those evaluated under Alternative 1. The radionuclide concentration in this salt would be about one-third of those generated in Alternative 1.

Table 4–7 Annual Cancer Risks Due to Accidents Under the No Action Alternative

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Salt powder spill in Hot Fuel Examination Facility cell	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Cask drop during spent nuclear fuel transfer	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Beyond-design-basis earthquake	4.8×10^{-10}	5.5×10^{-8}	1.5×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 4.8×10^{-8} per year (or one chance in 20.8 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 5.6×10^{-6} per year (or one chance in 178,600 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological hazardous chemical impacts are evaluated in terms of comparison to ERPGs. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The nonradiological (hazardous chemical) impacts of potential facility accidents associated with either option of the No Action Alternative are summarized in **Table 4–8**.

Table 4–8 Hazardous Chemical Accident Impacts Under the No Action Alternative

Accident	*Frequency (Event Per Year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1 Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1 Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.2.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.2.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the spent nuclear fuel storage facilities at ANL-W and INTEC to be much lower than 1. Therefore, there would be no disproportionately high and adverse consequences for any particular group within the general population, including minority or low-income populations, beyond the effects of existing and future activities that are independent of the proposed action.

4.2.6 Waste Management

Various types of waste would be generated as a result of sodium-bonded spent nuclear fuel storage activities at ANL-W, including transuranic waste, low-level radioactive waste, mixed waste, hazardous, and nonhazardous waste. In addition, during the first two years of operation under either option of this alternative, ANL-W would continue to generate high-level radioactive waste as the Electrometallurgical Treatment Research and Demonstration Project cladding hull waste and electrorefiner salt are stabilized to metallic and ceramic high-level radioactive waste forms for ultimate disposal. **Table 4–9** shows the anticipated categorization of these waste types and their expected interim storage and final disposal locations. The quantities of ceramic and metallic waste forms generated, along with other generated waste, are presented in **Table 4–10**. The values in Table 4–10 are for disposal (solid waste) and account for volume reduction.

Direct Process Waste

Under either option of the No Action Alternative, small amounts of metallic and ceramic high-level radioactive waste would be produced at ANL-W as a result of the completion of the demonstration project. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. Both the ceramic and metallic waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–10 are for the standardized canisters required for disposal of these materials.

Table 4–9 Waste Material Categories at INEEL and Interim and Final Locations

Waste Stream	*Category*	*Interim Storage Location*	*Final Disposal Location*
Process Waste			
Fuel hardware	Low-level radioactive waste	None	Radioactive Waste Management Complex
Metallic waste form	High-level radioactive waste	Radioactive Scrap and Waste Facility	Geologic repository
Ceramic waste form	High-level radioactive waste	Radioactive Scrap and Waste Facility	Geologic repository
Other Associated Process Waste			
Less than 10 nanocuries per gram transuranic waste [a]	Low-level radioactive waste	None	Radioactive Waste Management Complex
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Radioactive Waste Management Complex	Waste Isolation Pilot Plant
Cadmium-contaminated	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant or Radioactive Waste Management Complex after treatment
Nonradioactive	Sanitary waste	None	INEEL landfill
Deactivation Waste			
Electrorefiner cadmium	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant
Equipment less than 10 nanocuries per gram transuranic waste [a]	Low-level radioactive waste	None	Radioactive Waste Management Complex
Equipment greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Radioactive Waste Management Complex	Waste Isolation Pilot Plant
Cadmium-contaminated	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant or Radioactive Waste Management Complex

[a] As noted in Section 3.2.11.3, the Radioactive Waste Management Complex cannot be used for the disposal of the alpha low-level radioactive waste (between 10 and 100 nanocuries per gram). Waste in this category may be treated by the Advanced Mixed Waste Treatment Project and then disposed of at the Waste Isolation Pilot Plant.

The metallic and ceramic high-level radioactive waste generated as a result of the demonstration project at ANL-W would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W in a manner that allows retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste, and shielding will be provided by a combination of: (1) steel storage liners in which the waste would be stored and (2) by the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, packaged in standardized canisters, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository. If direct disposal of sodium-bonded spent nuclear fuel becomes acceptable, the sodium-bonded spent nuclear fuel at INTEC would be transferred to ANL-W for repackaging, along with other fuel at ANL-W. The packaged canisters would be transferred to the INEEL Dry Transfer Facility for shipment off site to a repository.

Table 4–10 Amounts of Waste Generated Under the No Action Alternative[a]

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	9.4 (15 canisters) [b]	14,000
High-level radioactive metallic waste	0.6 (1 canister) [b]	460
Spent nuclear fuel	142 (355 canisters) [b]	72,000
Other Associated Process Waste		
Low-level radioactive waste	792	161,000
Transuranic waste	10.5	4,000
Mixed waste	40	21,500
Sanitary waste	2,500	867,000
Deactivation Waste		
Low-level radioactive waste	112	38,000
Transuranic waste	1.6	853
Mixed waste	3	2,100

[a] These waste generation estimates are through the year 2035. This is the date by which materials of this type are required to be out of the State of Idaho.
[b] Standardized canisters.
Source: ANL 1999.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated during conversion of demonstration high-level radioactive waste into suitable forms for the repository, as well as from other ongoing activities, including keeping a hot cell facility operational to handle unforeseen problems while storing the sodium-bonded spent nuclear fuel at INTEC or in the Radioactive Scrap and Waste Facility. In addition, low-level radioactive waste would be generated from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from either option of the No Action Alternative activities at ANL-W that would require disposal (after volume reduction) would be a maximum of about 50 cubic meters (1,766 cubic feet) per year during processing activities, and approximately 17 cubic meters (600 cubic feet) per year during the remaining years. This maximum volume represents a small fraction (approximately 1 percent) of the total annual volume of low-level radioactive waste currently being disposed of at the Radioactive Waste Management Complex. The total of 792 cubic meters (28,000 cubic feet) of low-level radioactive waste generated during either option represents approximately 0.7 percent of the total Radioactive Waste Management Complex disposal inventory.

Other Associated Process Transuranic Waste

Transuranic waste would be generated at ANL-W under either option of the No Action Alternative from decontamination activities for repair and maintenance of items, and miscellaneous work associated with

demonstration fuel processing or other activities. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

For the No Action Alternative, the volume of transuranic waste generated at ANL-W would amount to a maximum of approximately 1 cubic meter (35 cubic feet) per year during processing activities, and approximately 0.2 cubic meters (7 cubic feet) per year during the remaining years. This maximum volume is approximately 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex. The total volume of incidental transuranic waste generated under either option is approximately 10.5 cubic meters (370 cubic feet), which is 0.006 percent of the estimated total volume of transuranic waste to be emplaced at the Waste Isolation Pilot Plant.

Other Associated Process Sanitary Waste

Sanitary waste, which is nonradioactive and nonhazardous solid waste, would continue to be generated under either option of the No Action Alternative. This waste would be typical of industrial operations and would be disposed of at the INEEL landfill. Based on an estimated eventual INEEL landfill volume of 3×10^6 cubic meters (106 million cubic feet), the total volume of sanitary waste generated and disposed of under this alternative is approximately 0.1 percent of the INEEL landfill volume.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or clean-up material and the analysis of cadmium samples. At ANL-W, mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated at ANL-W have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities at ANL-W. This would include process equipment and process material such as cadmium in one of the electrorefiners. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste under either option of the No Action Alternative would be low-level radioactive waste, generated as a result of equipment dismantling and disposal. Components that would require disposal include the existing electrorefiners and hot isostatic press, as well as other processing components. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated over a period of one year. The total deactivation waste represents approximately 14 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

4.3 ALTERNATIVE 1: ELECTROMETALLURGICALLY TREAT BLANKET AND DRIVER FUEL AT ANL-W

Under this alternative, the sodium-bonded spent nuclear fuel would be treated at ANL-W using the electrometallurgical process, described in Appendix C. The various process steps in this technology are performed at the Fuel Conditioning Facility and the Hot Fuel Examination Facility hot (air or argon) cells. The

processes at the Fuel Conditioning Facility include: fuel chopping, electrorefining, cathode processing, and metal casting (see Appendix C for details on each processing step). These processes would separate the uranium from the fission products. Separated uranium is not considered a waste. The separated uranium would be made into low-enriched uranium ingots, and the metallic sodium would be oxidized in the electrorefiner lithium-potassium salt and removed along with the fission products as high-level radioactive waste. The salts from the electrorefiner then would be solidified and sent to the Hot Fuel Examination Facility for further processing. The processes at the Hot Fuel Examination Facility include waste treatment, metallic melting, and high-level radioactive waste production. These processes would produce two waste forms—a ceramic waste form consisting of fission products and transuranic elements and a metallic waste form consisting of noble metal fission products and cladding hulls from the spent nuclear fuel. The low-enriched uranium metal ingots would be stored at the Zero Power Physics Reactor Material Storage Building. The ceramic and metallic waste forms would be temporarily stored at the Radioactive Scrap and Waste Facility pending packaging for disposition in a geologic repository.

The electrometallurgical process at ANL-W facilities would treat about 5 metric tons of heavy metal of sodium-bonded spent nuclear fuel per year. Appendix E, Section E.4.1, provides details on the process duration and the amount of blanket and driver spent nuclear fuel treated annually. The treatment of blanket and driver spent nuclear fuel under this alternative could start as early as 2000 and could be completed by 2012. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.3.1 Air Quality

Nonradiological Gaseous Emissions

As explained in Appendix E, Section E.5.3.1, under all alternatives, small quantities of criteria and hazardous chemicals are generated from the operation of the emergency diesel generators supporting both the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W. The emissions from these generators are independent of any of the treatment processes addressed in this EIS. In addition, the electrometallurgical treatment of driver fuel under Alternatives 1 through 5 would release small quantities of cadmium. This release would occur as an elevated release from the Fuel Conditioning Facility stack.

Table 4–11 summarizes the concentrations of criteria and hazardous air pollutants. The concentrations are compared to their corresponding ambient air quality standards. Only those air pollutants that are expected and have ambient air quality standards are presented in the table. The emissions are generated from diesel generators currently in operation and are considered as part of the baseline concentration. No increases in emissions are expected under this alternative. Therefore, a Prevention of Significant Deterioration increment analysis was not required. In addition, the INEEL site is located in areas of attainment for criteria pollutants; therefore, no conformity analysis is required.

Radiological Gaseous Emissions

Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the argon cell at the Fuel Conditioning Facility during fuel element chopping and electrorefining processes. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell, and releases of tritium to the atmosphere would be in the elemental form. The oxidation of elemental tritium to tritium oxide (HTO or T_2O) has been shown to occur slowly in the environment, and in the long term, about 1 percent of tritium would be oxidized (see Appendix E, Section E.4.1, for more details). The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries

(10^{-9} curies) and are released to the environment through the facility stack, along with krypton and elemental tritium. The maximum release of radioactive gaseous emissions would occur during the first six years of the electrometallurgical treatment process, when a combination of EBR-II blanket and driver spent nuclear fuel elements would be processed. During these six years, about 0.6 metric tons of heavy metal of driver spent nuclear fuel and about 4.4 metric tons of heavy metal of blanket spent nuclear fuel would be processed annually. The combined process would release about 11,600 curies of krypton-85 and 770 curies of elemental tritium annually. After six years and until the end of the processing period, the release rate would drop significantly. During this period, only Fermi-1 blanket spent nuclear fuel, with an annual release of about 0.4 curies of elemental tritium and 3.3 curies of krypton-85, would be processed. The radiological exposures to the public and workers from these emissions are presented in detail in Appendix E, Section E.4.1, and are summarized in Section 4.3.4.

Table 4–11 Nonradiological Air Quality Concentrations at the Site Boundary Under Alternative 1 at ANL-W for Comparison With Ambient Air Quality Standards

	Averaging Period	*Most Stringent Standard or Guideline (micrograms per cubic meter)* [a]	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutant			
Carbon monoxide	8 hours 1 hour	10,000 40,000	32.70 46.80
Nitrogen dioxide	Annual	100	2.79
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 150 150	0.01 0.19 Not available
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.45 11.50 25.80
Hazardous and Toxic Compounds			
1,3-Butadiene	Annual	0.0036	0.0000355
Acetaldehyde	Annual	0.45	0.0000226
Acrolein	24 hours	12.5	0.000181
Benzene	Annual	0.12	0.000694
Cadmium	Annual	0.00056	3.58×10^{-10}
Formaldehyde	Annual	0.077	0.0000709
Toluene	24 hours	18,750	0.00664
Xylene	24 hours	21,750	0.00447

PM_n = Particulate matter less than or equal to *n* microns in diameter.

[a] The standards for hazardous and toxic compounds apply only to increases in emissions from new or modified sources and are provided for information purposes only, as concentrations from releases at ANL-W under all alternatives are not expected to increase.

4.3.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with the electrometallurgical treatment processes. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W (see Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent or waste generated by the electrometallurgical treatment process would be discharged to surface water.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. The current water use at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by the electrometallurgical treatment process would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent generated by the electrometallurgical treatment process would be discharged to groundwater.

4.3.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

4.3.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–12 and 4–13 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–15 and 4–16. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–17. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.3.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during fuel chopping and from the operation of electrorefiners. Both of these activities are performed in the Fuel Conditioning Facility argon cell. Appendix E, Sections E.3 and E.4.1, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result from treating 0.6 metric tons of heavy metal of EBR-II driver spent nuclear fuel and 4.4 metric tons of heavy metal of EBR-II blanket spent nuclear fuel. This combination of fuel treatment would continue for 6 years, after which only Fermi-1 blanket spent nuclear fuel with a very low radioactivity content would be treated. Overall, it would require 13 years to treat all the sodium-bonded fuel (see Appendix E, Section E.4.1, for details).

Calculated maximum annual and project total radiological impacts to the public from operational activities under this alternative are given in **Table 4–12**. The maximum dose to the public would occur during the first six years of operation. The annual dose to the public during Fermi-1 blanket spent nuclear fuel treatment would be very small (see Appendix E, Table E–8). The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). As explained in Appendix E, Section E.4.1, the dose resulting from the release of tritium depends heavily on the chemical form. The inhalation dose from oxidized tritium (HTO or T_2O) is 25,000 times higher than for elemental tritium (HT or T_2). In the environment, about 1 percent of elemental tritium would be oxidized over the long term. In this analysis, about 1 percent of the tritium conservatively was assumed to be in oxidized form at the time of release.

Table 4–12 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 1

Receptor	*Electrometallurgically Treat Driver Spent Nuclear Fuel*	*Electrometallurgically Treat Blanket Spent Nuclear Fuel*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem per year) [a]	0.0027	0.000083	0.0028
Excess latent cancer fatalities (per year)	1.4×10^{-6}	4.2×10^{-8}	1.4×10^{-6}
Project total excess latent cancer fatalities [b]	8.0×10^{-6}	2.2×10^{-7}	8.2×10^{-6}
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.00033	0.000010	0.00034
Percent of annual background radiation [c]	0.000092	2.8×10^{-6}	0.000094
Latent cancer fatality risk (per year)	1.6×10^{-10}	5.0×10^{-12}	1.7×10^{-10}
Project total lifetime cancer fatality risk [b]	9.6×10^{-10}	2.6×10^{-11}	9.9×10^{-10}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.000011	3.5×10^{-7}	0.000012
Latent cancer fatality risk (per year)	5.6×10^{-12}	1.7×10^{-13}	5.8×10^{-12}
Project total lifetime cancer fatality risk [b]	3.3×10^{-11}	9.1×10^{-13}	3.4×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over 13 years of emissions.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Primary contributors to doses to members of the public are from releases of tritium gas and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in Table 4–12. As shown in the table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with the electrometallurgical treatment of sodium-bonded spent nuclear fuel. The estimated annual worker population collective dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 13 years of electrometallurgical treatment activities, and a 1-year dose (33 person-rem) from deactivation activities is included, the project total worker population dose would be 319 person-rem, leading to a risk of 0.13 latent cancer fatalities (see **Table 4–13**).

Table 4–13 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 1

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk project total over 13 years	0.00031
Worker Population	
Collective dose (person-rem per year)	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	319
Project total excess latent cancer fatalities	0.13

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year as established for all DOE activities in DOE Order N 441.1.

Source: ANL 1999.

As shown in Tables 4–12 and 4–13:

- The maximum dose to the maximally exposed offsite individual would be 0.00034 millirem per year, with an associated risk of developing a fatal cancer of 1.7×10^{-10} per year (or one chance in 5.9 billion that the individual would develop a fatal cancer per year of exposure).

- The maximum collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0028 person-rem per year, with an associated 1.4×10^{-6} latent cancer fatalities per year (or one chance in 667,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.2×10^{-6} latent cancer fatalities (or one chance in 122,000 that the exposed population would experience a fatal cancer).

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 319 person-rem, with an associated 0.13 latent cancer fatalities (or one chance in seven that the workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under this alternative are summarized in **Table 4–14**. Appendix E, Section E.5, provides details on the model used and results obtained. The results, presented in Table 4–14, indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

Table 4–14 Hazardous Chemical Impacts to the Public From Operational Activities Under Alternative 1

Chemical	*Maximum Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
1,3-Butadiene	3.6×10^{-8}	None	9.9×10^{-9}
Acetaldehyde	2.3×10^{-8}	2.5×10^{-6}	5.0×10^{-11}
Acrolein	7.1×10^{-9}	0.00035	None
Benzene	6.9×10^{-7}	None	5.4×10^{-9}
Cadmium	3.6×10^{-13}	None	6.5×10^{-13}
Formaldehyde	7.1×10^{-8}	None	9.2×10^{-10}
Toluene	2.5×10^{-7}	6.2×10^{-7}	None
Hazard Index		0.00036	Not applicable

4.3.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during electrometallurgical treatment operational activities are summarized and presented in this section. Since electrometallurgical treatment processes are performed in both the Fuel Conditioning Facility and the Hot Fuel Examination Facility, accidents at both facilities would be considered. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios including fire, spills, criticality, earthquake, and aircraft crash. Aircraft crash and criticality accidents were determined to have an accident frequency of less than 10^{-7} per year, and were not analyzed further. **Table 4–15** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents

(see discussions on the involved worker in Section 4.2.4.2). The accident risks are summarized in **Table 4–16**.

Table 4–15 Accident Frequency and Consequences Under Alternative 1

Accident	*Frequency (event per year)*	*Maximally Exposed Offsite Individual*		*Population Within 80 Kilometers (50 Miles)*		*Noninvolved Worker*	
		Dose (millirem)	*Latent Cancer Fatality Risk* [a]	*Dose (person-rem)*	*Excess Latent Cancer Fatalities* [b]	*Dose (millirem)*	*Latent Cancer Fatality Risk* [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Salt powder spill	0.01	0.00015	7.5×10^{-11}	0.000033	1.7×10^{-8}	1.3×10^{-6}	5.3×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.065	3.3×10^{-8}	0.0077	3.9×10^{-6}	0.22	8.8×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Design-basis earthquake	0.008	4.0	2.0×10^{-6}	0.47	0.00024	14	5.6×10^{-6}
Beyond-design-basis earthquake	0.00001	930	0.00047	110	0.055	560	0.00023

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Table 4–16 Annual Cancer Risks Due to Accidents Under Alternative 1

Accident	*Maximally Exposed Offsite Individual* [a]	*Population Within 80 Kilometers (50 Miles)* [b]	*Noninvolved Worker* [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Salt powder spill	7.5×10^{-13}	1.7×10^{-10}	5.3×10^{-15}
Salt transfer drop	3.3×10^{-15}	3.9×10^{-13}	8.8×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Design-basis earthquake	1.6×10^{-8}	1.9×10^{-6}	4.5×10^{-8}
Beyond-design-basis earthquake	4.7×10^{-9}	5.5×10^{-7}	2.3×10^{-9}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 4.5×10^{-8} per year (or one chance in 22.2 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPGs. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details). The nonradiological impacts of potential facility accidents associated with the electrometallurgical treatment alternative at ANL-W are summarized in **Table 4–17**.

Table 4–17 Hazardous Chemical Accident Impacts Under Alternative 1

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.3.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.3.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.3.6 Waste Management

Electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated

categorization of waste types and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of electrometallurgical treatment at ANL-W are provided in **Table 4–18**.

Table 4–18 Amounts of Waste Generated Under Alternative 1[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	12.5	6,600
High-level radioactive ceramic waste	78 (125 canisters) [b]	120,000
High-level radioactive metallic waste	3.1 (5 canisters) [b]	9,000
Other Associated Process Waste		
Low-level radioactive waste [c]	706	143,000
Transuranic waste	12.5	5,400
Mixed waste	35.3	19,000
Sanitary waste	4,960	1.72×10^6
Deactivation Waste		
Low-level radioactive waste [c]	143	48,000
Transuranic waste	1.6	853
Mixed waste	4.2	2,900

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.
[b] Standardized canisters.
[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.
Source: ANL 1999.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that account only for the fraction of total ANL-W waste that would be attributable to the processing of sodium-bonded spent nuclear fuel under this alternative. The values in Table 4–18 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of electrometallurgical treatment could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

The waste values in Table 4–18 are total quantities that would be generated as a result of Alternative 1 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 1, the sodium-bonded spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For electrometallurgical treatment, fuel assembly hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been

produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures.

Under Alternative 1, metallic and ceramic high-level radioactive waste would be a primary product. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–18 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste generated would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from electrometallurgical treatment at ANL-W that would require disposal (after volume reduction) would be approximately 48 cubic meters (1,695 cubic feet) per year. This represents approximately 0.08 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 706 cubic meters (24,932 cubic feet) represents approximately 0.9 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated by decontamination activities for repair and maintenance of items, and miscellaneous work associated with the electrometallurgical processing. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is 12.5 cubic meters (441 cubic feet), which is less than 0.008 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste of this category would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with electrometallurgical treatment processing at ANL-W. This would include process equipment and process material, such as electrorefiner cadmium. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste, transuranic waste, and mixed waste generated as a result of equipment dismantling and disposal. Components that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated in a single year. This waste would represent an increase of approximately 3.5 times the annual waste generated by electrometallurgical treatment requiring disposal. The total deactivation waste would represent approximately 20 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

4.4 Alternative 2: Clean and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be cleaned to remove metallic sodium and placed in high-integrity cans. These cans then would be placed into overpack containers prior to dry storage at the Radioactive Scrap and Waste Facility, pending repackaging and transportation for disposal in a geologic repository. The removed sodium contains radioactive elements, principally cesium. The cesium would be separated from the sodium and stabilized as ceramic waste. The sodium would be stabilized using an oxidation/carbonation process (see Appendix C for more detail) (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its placement in high-integrity cans could start in 2003 and could be completed by 2009. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.4.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail).

Radiological Gaseous Emissions

The cleaning of the blanket spent nuclear fuel to remove metallic sodium and the electrometallurgical treatment of the driver spent nuclear fuel would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. Appendix E, Section E.4.2, provides details on releases during the processing period at ANL-W. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions would occur when cleaning blanket spent nuclear fuel for placement in high-integrity cans and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. This simultaneous operation was estimated to occur over a three-year period starting in 2003. Based on an annual cleaning throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel and an electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere. The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.4.4.

4.4.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with high-integrity can operations and electrometallurgical treatment process operations. Current operating and monitoring practices would continue for NPDES stormwater and liquid effluent discharges associated with facilities at ANL-W (see also Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent or waste generated by high-integrity can and electrometallurgical treatment process operations would be discharged to surface water.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. The current water usage at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by high-integrity can and electrometallurgical treatment process operations would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent or waste generated by high-integrity can and electrometallurgical treatment process operations would be discharged to groundwater.

4.4.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

4.4.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–19 and 4–20 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–21 and 4–22. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–23. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.4.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel cleaning and driver spent nuclear fuel chopping. All of these activities would be performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when cleaning of blanket spent nuclear fuel and treatment of driver spent nuclear fuel are performed simultaneously under this alternative. Appendix E, Section E.4.2, provides details on the treatment process duration and throughputs for each fuel type. The duration of the treatment process is estimated to be nine years.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–19**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public are releases of tritium gas (about 1 percent of which was conservatively assumed to be in oxidized form) and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–19 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 2

Receptor	*Electrometallurgically Treat Driver Spent Nuclear Fuel*	*Clean and Place Blanket Spent Nuclear Fuel in High-Integrity Cans*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem per year) [a]	0.0027	0.00028	0.0030
Excess latent cancer fatalities (per year)	1.4×10^{-6}	1.4×10^{-7}	1.5×10^{-6}
Project total excess latent cancer fatalities [b]	8.0×10^{-6}	3.4×10^{-7}	8.3×10^{-6}
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.00033	0.000048	0.00038
Percent of annual background [c]	0.000092	0.000013	0.00011
Latent cancer fatality risk (per year)	1.7×10^{-10}	2.4×10^{-11}	1.9×10^{-10}
Project total lifetime cancer fatality risk	9.4×10^{-10}	5.8×10^{-11}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.000011	1.2×10^{-6}	0.000012
Latent cancer fatality risk (per year)	5.6×10^{-12}	5.8×10^{-13}	6.2×10^{-12}
Project total lifetime cancer fatality risk [b]	3.3×10^{-11}	1.4×10^{-12}	3.5×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over nine years.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–20 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with sodium-bonded spent nuclear fuel high-integrity can and electrometallurgical treatment processes. It was concluded that the average worker dose would not be different from that currently being experienced. The estimated annual collective worker dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the nine years of treatment activities, plus one year for deactivation of the facilities, the project total worker population dose would be 231 person-rem, leading to a risk of 0.092 latent cancer fatalities.

As shown in Tables 4–19 and 4–20:

- The annual dose to the maximally exposed offsite individual would be 0.00038 millirem per year, with an associated 1.9×10^{-10} risk per year of developing a fatal cancer (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 666,700 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

Table 4–20 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 2

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over nine years)	0.00022
Worker Population	
Collective dose (person-rem per year) [b]	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	231
Project total excess latent cancer fatalities	0.092

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

[b] Worker dose is 33 person-rem for one year of deactivation activities.

Source: ANL 1999.

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 231 person rem, with an associated 0.092 latent cancer fatalities (or one chance in 11 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 2 would be similar to the impacts evaluated for Alternative 1 described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is described in Section 3.2.10.2.

4.4.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during cleaning activities for placement of blanket spent nuclear fuel elements in high-integrity cans and the electrometallurgical treatment operational activities for driver spent nuclear fuel are summarized and presented in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F.

The detailed analysis considered a wide spectrum of potential accident scenarios including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for these two events were not

performed. Cleaning of the blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of the driver spent nuclear fuel would be performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event. The cleaning of the blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. **Table 4–21** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility.

Table 4–21 Accident Frequency and Consequences at ANL-W Under Alternative 2

Accident	*Frequency (event per year)*	*Maximally Exposed Offsite Individual*		*Population Within 80 Kilometers (50 Miles)*		*Noninvolved Worker*	
		Dose (millirem)	*Latent Cancer Fatality Risk* [a]	*Dose (person-rem)*	*Excess Latent Cancer Fatalities* [b]	*Dose (millirem)*	*Latent Cancer Fatality Risk* [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this accident is the frequency for the facility design-basis earthquake initiating cell fire.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–22**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would develop a fatal cancer per year of operation).

Table 4–22 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 2

Accident	*Maximally Exposed Offsite Individual* [a]	*Population Within 80 Kilometers (50 Miles)* [b]	*Noninvolved Worker* [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The hazardous chemical impacts of potential facility accidents associated with the treatment of driver spent nuclear fuel using the electrometallurgical treatment process are summarized in **Table 4–23**.

Table 4–23 Hazardous Chemical Impacts Due to Accidents at ANL-W Under Alternative 2

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.008	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.4.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.4.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical treatment processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.4.6 Waste Management

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel assembly hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 2 are provided in **Table 4–24**.

Table 4–24 Amounts of Waste Generated Under Alternative 2[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	12.5	6,000
High-level radioactive ceramic waste	16.3 (26 canisters) [b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters) [b]	2,500
Spent nuclear fuel	25.2 (63 canisters) [b]	63,000
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister) [b]	220
Low-level radioactive waste [c]	555	113,000
Transuranic waste	9.1	3,800
Mixed waste	27.5	14,800
Sanitary waste	4,960	1.72×10^6
Deactivation Waste		
Low-level radioactive waste [c]	166.2	56,000
Transuranic waste	1.6	853
Mixed waste	4.8	3,200

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–24 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 2 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

The waste values in Table 4–24 are total quantities that would be generated as a result of Alternative 2 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 2, the driver spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) for disposal in the repository. The blanket spent nuclear fuel would be cleaned and packaged in high-integrity cans for disposal in the repository. In this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste would be categorized as high-level radioactive waste.

The packaged spent nuclear fuel volume is based on placing the blanket spent nuclear fuel in high-integrity cans which would be placed in standardized canisters. The volumes of waste forms provided in Table 4–24 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel under this alternative, driver and blanket spent nuclear fuel generated during the demonstration project, and packaged spent nuclear fuel would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing these materials would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste could be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste

stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of processing at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area in the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.5 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated by Alternative 2 from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated as a result of the treatment of sodium-bonded spent nuclear fuel at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is 9.1 cubic meters (321 cubic feet), which is 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with electrometallurgical treatment processing at ANL-W. This would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste

categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated in two years. The total deactivation waste would represent an additional 30 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

4.5 ALTERNATIVE 3: DECLAD AND CLEAN BLANKET FUEL AND ELECTROMETALLURGICALLY TREAT DRIVER FUEL AT ANL-W; PUREX PROCESS BLANKET FUEL AT SRS

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be declad and cleaned to remove metallic sodium, packaged in aluminum cans at ANL-W, and shipped to SRS for treatment using the PUREX process at F-Canyon. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The high-level radioactive waste generated from the treatment of the blanket spent nuclear fuel at SRS would be in the form of borosilicate glass and would be stored at the SRS Defense Waste Processing Facility, pending repackaging and transportation for disposal in a geologic repository. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its shipment to SRS could start in 2003 and could be completed by 2009. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

PUREX processing of blanket spent nuclear fuel at SRS would require six months of operation and could be completed by 2010.

4.5.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative to be the same as for Alternative 1, as described in Section 4.3.1, (see also Appendix E, Section E.5.3.1 for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

The concentrations of nonradiological air pollutants attributed to this alternative at SRS are presented in **Table 4–25**. These concentrations are based on information in the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (SRS Spent Nuclear Fuel Management Final EIS) (DOE 2000) for the PUREX processing of similar fuel. See Appendix E, Section E.5.3.2, for more details. The site boundary concentrations are equal to the incremental concentrations generated in this alternative plus the baseline concentrations given in Section 3.3.3.1. Only those air pollutants that are expected and have ambient air quality standards are presented in the table. Note that there are no Prevention of Significant Deterioration increment-consuming sources at SRS; therefore, a Prevention of Significant Deterioration increment analysis was not performed.

Table 4–25 Nonradiological Air Quality Concentrations at the Site Boundary Under Alternative 3 at SRS for Comparison With Ambient Air Quality Standards

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutants			
Carbon monoxide	8 hours 1 hour	10,000 40,000	1.22 9.06
Nitrogen dioxide	Annual	100	3.11
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 150 150	Less than 0.01 0.11 Not available
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	Less than 0.01 0.12 0.91
State-regulated Pollutants			
Gaseous fluoride	30 days 7 days 24 hours 12 hours	0.8 1.6 2.9 3.7	0.01 0.03 0.06 0.11
Total suspended particulates	Annual	75	Less than 0.01
Hazardous/Toxic Compounds			
1,1,1-trichloroethane	24 hours	9,550	Less than 0.01
Benzene	24 hours	150	0.01
Ethanolamine	24 hours	200	Less than 0.01
Ethyl benzene	24 hours	4,350	Less than 0.01
Ethylene glycol	24 hours	650	Less than 0.01
Formaldehyde	24 hours	15	Less than 0.01
Glycol ethers	24 hours	No standard	Less than 0.01
Hexachloronaphthalene	24 hours	1	Less than 0.01
Hexane	24 hours	900	0.01
Manganese	24 hours	25	Less than 0.01
Methyl alcohol	24 hours	1,310	Less than 0.01
Methyl-ethyl-ketone	24 hours	14,750	Less than 0.01
Methyl-isobutyl-ketone	24 hours	2,050	Less than 0.01
Methylene chloride	24 hours	8,750	0.01
Naphthalene	24 hours	1,250	Less than 0.01
Nitric acid	24 hours	125	0.28
Phenol	24 hours	190	Less than 0.01
Phosphorous	24 hours	0.5	Less than 0.01
Sodium hydroxide	24 hours	50	Less than 0.01
Toluene	24 hours	2,000	0.01
Trichloroethane	24 hours	6,750	Less than 0.01
Vinyl acetate	24 hours	176	Less than 0.01
Xylene	24 hours	4,350	0.02

PM_n = Particulate matter less than or equal to *n* microns in diameter.
Source: Bickford et al. 1997.

Radiological Gaseous Emissions

The decladding and cleaning of blanket spent nuclear fuel and the electrometallurgical treatment of driver spent nuclear fuel at ANL-W would release gaseous fission products to the hot (argon) cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with the krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions would occur when decladding blanket spent nuclear fuel for packaging and shipment to SRS and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. This simultaneous operation was estimated to occur over a three-year period starting in 2003. Appendix E, Section E.4.2, provides details on releases during the processing period at ANL-W. Based on an annual decladding throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and an electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere (see Appendix E, Section E.4.2).

Since declad and clean fuel would be packaged and sent to SRS, some gaseous fission products would be expected to be present in that fuel. However, it was assumed conservatively that all gaseous fission products in the blanket spent nuclear fuel would be released to the environment during PUREX processing at SRS over a six-month period (see Appendix E, Section E.4.3). The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.5.4.

4.5.2 Water Resources

As stated in Section 4.4.2, decladding and cleaning of blanket spent nuclear fuel and treatment of driver spent nuclear fuel using electrometallurgical treatment would not discharge any radiological chemical material to the surface or groundwater at the INEEL site. These activities also would not impact the current groundwater usage at the site. For a discussion of impacts on water resources at ANL-W, see Section 4.4.2.

The impacts on water resources from processing blanket spent nuclear fuel at F-Canyon are described below.

Surface Water

No surface water would be used for PUREX processing of blanket spent nuclear fuel at the F-Area. The F-Canyon processing facilities are outside the 100-year floodplain, as shown in Figure 3-6.

Nonradiological Liquid Effluent

The major sources of liquid effluent from PUREX processing of blanket spent nuclear fuel at SRS would be process cooling water and steam condensate. There are sufficient capacities in existing wastewater treatment facilities to handle the liquid effluent from this processing. Liquid effluent associated with PUREX processes would use these facilities and the existing permitted outfalls (Section 3.3.4.1). Process cooling water treatment would result in releases to Upper Three Runs Creek from the F-Area, as shown in **Table 4–26**. Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility and discharged through an existing NPDES-permitted outfall. Since employment would not increase as a result of processing this fuel, the treatment rates through the Central Wastewater Treatment Facility would not be affected and the requirements of the SRS NPDES permit would continue to be met (DOE 2000).

Table 4–26 Chemical Effluent Concentrations From PUREX Cooling Water Treatment

Parameter	Effluent Concentrations	Existing Stream Water Concentrations		Water Quality Criterion (milligrams per liter) [c]
	F-Area (milligrams per liter)	Upper Three Runs (Upstream) [a] (Average) (milligrams per liter)	Upper Three Runs (Downstream) [b] (Average) (milligrams per liter)	
Aluminum	0.2	0.19	0.24	(d)
Ammonia	0.03	0.0001	Not reported	(d)
Chromium	0.02	Not detected	Not detected	0.1
Copper	0.01	0.018	0.015	1
Manganese	0.01	0.039	0.052	0.05
Nickel	0.05	Not detected	Not detected	0.1
Nitrate	0.04	0.36	0.27	10
Zinc	0.07	0.06	0.091	3

[a] Stream monitor U3R-1A.
[b] Stream monitor U3R-4.
[c] Federal Drinking Water Standards and Health Advisories (EPA 1996) and South Carolina Water Quality Criteria for Protection of Human Health (SCDHEC 1998).
[d] No drinking water standard.
Sources: Arnett and Mamatey 1998, DOE 2000.

Although proposed or final Federal drinking water standards do not apply to the discharges, these standards are used for comparison to SRS discharges. The discharge concentration would not exceed the Federal drinking water standard. The discharges would also comply with the South Carolina Water Classifications and Standards (SCDHEC 1998). The release concentrations would be no greater than the concentrations measured in Upper Three Runs (Arnett and Mamatey 1998), with the exception of zinc and ammonia. Zinc concentrations in the discharge are within the Federal health advisory limits (EPA 1996).

Radiological Liquid Effluent

PUREX processing would release measurable radioactive nuclides to the surface water through the cooling water system. The expected radiological effluent from processing declad and cleaned blanket spent nuclear fuel at F-Canyon was estimated based on the measured data from various effluent streams at F-Area as presented in the SRS Environmental Data for 1997 (Arnett and Mamatey 1998). Since the mechanism associated with releases of liquid effluent from PUREX processing at F-Canyon is essentially the same for almost every fuel type processed, the F-Area 1997 effluent data were used to conservatively represent the potential releases from a six-month operation of F-Canyon. **Table 4–27** provides a list of potential radiological isotopes that could be released to the surface water during processing of approximately 57 metric tons of heavy metal of blanket spent nuclear fuel (see Appendix E, Section E.4.3, for details).

Groundwater

All process water would come from groundwater, as would sanitary water. At most, less than 65 million liters (17 million gallons) per year would be required for cooling water. SRS annually withdraws more than 5 billion liters (more than 1.3 billion gallons) per year of groundwater (DOE 2000).

Table 4–27 Estimated Radiological Liquid Effluent From PUREX Processing of Blanket Spent Nuclear Fuel

Isotope	*Curies*
Tritium (Hydrogen-3)	1.54
Strontium-89/Strontium-90	0.000031
Cesium-137	0.0022
Uranium-234	0.000085
Promethium-147	0.000011
Uranium-238	0.00019
Plutonium-238	0.000016
Plutonium-239	7.8×10^{-6}

Source: Arnett and Mamatey 1998.

Nonradiological Liquid Effluent

No nonradiological chemicals would be discharged to groundwater from PUREX processing of blanket spent nuclear fuel at F-Canyon and the FB-Line in F-Area.

Radiological Liquid Effluent

No radiological liquid effluent or waste would be discharged to groundwater from PUREX processing of blanket spent nuclear fuel at F-Canyon and the FB-Line in F-Area.

4.5.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W and SRS would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the regions around INEEL and SRS.

4.5.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–28 and 4–29 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–31 through 4–34. The impacts from hazardous chemical releases during accident conditions are presented in Tables 4–35 and 4–36. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.5.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during PUREX processing at F-Canyon. Appendix E, Sections E.3 and E.4.3, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. Doses to the public would result from treating about 57 metric tons of heavy metal of blanket spent nuclear fuel. The blanket spent nuclear fuel being processed at SRS is already declad and cleaned at ANL-W; therefore, the gaseous fission products are assumed to have already been released. However, for the analytical purposes of this EIS, it was

conservatively assumed that the gaseous fission products would still be within the matrix of the fuel and would be released during PUREX processing at SRS. The processing was assumed to continue for six months (see Appendix E.4.3).

Calculated incremental maximum annual and project total radiological impacts to the public are given in **Table 4–28**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of F-Canyon in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the SRS site boundary and receiving the maximum dose). Since PUREX processing would produce radiological air emissions as well as radiological liquid effluent, doses to the public were calculated considering both the air emissions and liquid effluent. Primary contributors to public doses would be from tritium gas (assumed to be tritium oxide) and krypton-85, which together contribute over 95 percent of the total calculated doses. The doses resulting from liquid effluent were estimated from data provided in support of the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) (see Appendix E, Section E.4.3, for details). The doses and duration from decladding and cleaning blanket spent nuclear fuel and treatment of driver spent nuclear fuel at ANL-W would be similar to those presented for Alternative 2 in Section 4.4.4.1. To put the impacts into perspective, comparisons to natural background radiation levels are included in Table 4–28.

Table 4–28 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 3

Receptor	*PUREX Process Declad and Cleaned Blanket Spent Nuclear Fuel at SRS* [a, b, c]	*Declad and Clean Blanket Spent Nuclear Fuel and Electrometallurgically Treat Driver Spent Nuclear Fuel at ANL-W*
Population Within 80 Kilometers (50 Miles) in the Year 2010		
Collective dose (person-rem per year) [d]	0.020	0.0030
Excess latent cancer fatalities (per year)	0.000010	1.5×10^{-6}
Project total excess latent cancer fatalities [f]	0.000010	8.3×10^{-6}
Maximally Exposed Offsite Individual		
Dose (millirem per year) [d]	0.00051	0.00038
Percent of annual background radiation [e]	0.00017	0.00011
Latent cancer fatality risk (per year)	2.6×10^{-10}	1.9×10^{-10}
Project total lifetime cancer fatality risk [f]	2.6×10^{-10}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)		
Dose (millirem per year) [g]	0.000024	0.000012
Latent cancer fatality risk (per year)	1.2×10^{-11}	6.2×10^{-12}
Project total lifetime cancer fatality risk [f]	1.2×10^{-11}	3.5×10^{-11}

[a] Includes dose from air emissions and liquid effluent over the six-month processing duration.
[b] Doses to the population and the maximally exposed offsite individual from liquid effluent are 0.00068 person-rem and 0.00012 millirem, respectively.
[c] Since PUREX operations would last less than one year, the values of the project total dose and risk are equal to the corresponding annual values.
[d] Annual maximum dose during normal operations.
[e] The annual natural background radiation level at INEEL and at SRS is about 360 and 300 millirem, respectively, for the average individual (see Tables 3–8 and 3–20); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem at INEEL and 254,000 person-rem at SRS.
[f] Total calculated risk over nine years at ANL-W and six months at SRS.
[g] Obtained by dividing the population dose by the number of people projected to live in the year 2010 within 80 kilometers (50 miles) of ANL-W (240,338) and SRS F-Canyon (848,000).

As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public would be much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–29 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by SRS workers involved with the PUREX process. The estimated annual total worker population dose would be 75 person-rem, with an average individual dose of 500 millirem per year for each of the 150 involved workers. If these estimates were projected for six months of PUREX activities, the project total worker population dose would be 38 person-rem, leading to a risk of 0.015 latent cancer fatalities. The estimated annual total worker population dose to treat driver spent nuclear fuel at ANL-W is 22 person-rem, as indicated in Section 4.4.4.1.

Table 4–29 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 3

Receptor	*Impacts*	
Worker [a]	Operations at SRS	Operations at ANL-W
Average worker dose (millirem per year)	250 [b]	60
Average worker latent cancer fatality risk (project total)	0.00010 [b]	0.00022 [c]
Worker Population		
Collective dose (person-rem per year)	38 [b]	22
Excess latent cancer fatalities (per year)	0.015 [b]	0.0088
Project total dose (person-rem)	38 [b]	231 [c]
Project total excess latent cancer fatalities	0.015 [b]	0.092 [c]

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

[b] Operations at SRS to treat blanket spent nuclear fuel at F-Canyon are performed over six months.

[c] Operations at ANL-W to declad and clean blanket spent nuclear fuel and treat driver spent nuclear fuel are performed over nine years plus one year for deactivation of processing facilities; see Section 4.4.1.

Sources: ANL 1999, DOE 2000.

As shown in Tables 4–28 and 4–29:

- The annual dose to the maximally exposed offsite individual at ANL-W would be 0.00038 millirem per year, with an associated risk of developing a fatal cancer of 1.9×10^{-10} per year (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 670,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W site would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

- The collective dose to ANL-W facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to ANL-W facilities workers would be 231 person-rem, with an associated 0.092 latent cancer fatalities (or one chance in 11 that the exposed workers would experience a fatal cancer).

- The project total dose to the maximally exposed offsite individual at SRS from six-month PUREX processing would be 0.00051 millirem, with an associated risk of developing a fatal cancer of 2.6×10^{-10} (or one chance in 3.8 billion that the individual would develop a fatal cancer).

- The project total dose to the population within 80 kilometers (50 miles) of the F-Canyon would be 0.020 person-rem, with an associated 0.000010 latent cancer fatalities (or one chance in 100,000 that the exposed population would experience a fatal cancer).

- The project total dose to F-Canyon facility workers would be 38 person-rem, with an associated 0.015 latent cancer fatalities (or one chance in 67 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 3 would be similar to the impacts evaluated for Alternative 1 described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing baseline chemical environment is presented in Section 3.2.10.2.

For SRS, both carcinogenic and noncarcinogenic health effects to the public were assessed from exposure to hazardous chemicals; the results are summarized in **Table 4–30**.

Table 4–30 Hazardous Chemical Impacts to the Public From Operational Activities at SRS Under Alternative 3

Chemical	*Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
Benzene	1.4×10^{-6}	None	1.1×10^{-8}
Ethyl benzene	1.3×10^{-6}	1.3×10^{-6}	None
Formaldehyde	1.3×10^{-6}	None	1.6×10^{-8}
Hexane	1.4×10^{-6}	7.1×10^{-6}	None
Manganese	1.3×10^{-6}	0.025	None
Methyl ethyl ketone	2.5×10^{-6}	2.5×10^{-6}	None
Methylene chloride	7.1×10^{-7}	None	3.3×10^{-10}
Naphthalene	1.3×10^{-6}	0.00042	None
Toluene	1.4×10^{-6}	3.5×10^{-6}	None
Vinyl acetate	1.3×10^{-6}	6.3×10^{-6}	None
Hazard Index		0.025	Not applicable

The results indicate that no adverse toxic (noncarcinogenic) health effects or cancer potency are expected from exposure to hazardous chemicals released at SRS under this alternative. See Appendix E, Section E.5, for more details. The existing baseline chemical environment is presented in Section 3.3.10.2.

4.5.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities associated with decladding, cleaning, and PUREX processing of blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel are summarized and presented in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for these two events were not performed. Decladding and cleaning of blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of driver spent nuclear fuel would be performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event. Decladding and cleaning blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. The multifacility impacts of the beyond-design-basis earthquake are not relevant to the blanket spent nuclear fuel. Therefore, only the higher frequency design-basis earthquake event was analyzed for the blanket spent nuclear fuel. **Table 4–31** presents the frequencies and consequences of the postulated set of accidents at ANL-W to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility.

Table 4–31 Accident Frequency and Consequences at ANL-W Under Alternative 3

Accident	*Frequency (event per year)*	*Maximally Exposed Offsite Individual*		*Population Within 80 Kilometers (50 Miles)*		*Noninvolved Worker*	
		Dose (millirem)	*Latent Cancer Fatality Risk* [a]	*Dose (person-rem)*	*Excess Latent Cancer Fatalities* [b]	*Dose (millirem)*	*Latent Cancer Fatality Risk* [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this accident is the frequency for the facility design-basis earthquake initiating cell fire.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–32**.

Table 4–32 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 3

Accident	*Maximally Exposed Offsite Individual* [a]	*Population Within 80 Kilometers (50 Miles)* [b]	*Noninvolved Worker* [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would experience a fatal cancer per year of operation).

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during PUREX operational activities at SRS are summarized below. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. **Table 4–33** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 350 meters (1150 feet) from the facility. The 350-meter (1150-foot) distance leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the facility worker population dose due to accidents. The accident risks for the same receptors are summarized in **Table 4–34**.

Table 4–33 Accident Frequency and Consequences at SRS Under Alternative 3

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Fire (F-Canyon)	0.000061	610	0.00031	5500	2.8	2300	0.00092
Explosion (FB-Line)	0.00010	6.5	3.3×10^{-6}	53	0.027	19	7.6×10^{-6}
Design-basis earthquake (F-Canyon)	0.00013	1100	0.00055	2100	1.1	12000	0.0048
Design-basis earthquake (FB-Line)	0.00013	58	0.000029	120	0.06	900	0.00036
Criticality	0.00010	11	5.5×10^{-6}	59	0.030	37	0.000015

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Table 4–34 Annual Cancer Risks Due to Accidents at SRS Under Alternative 3

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Fire (F-Canyon)	1.9×10^{-8}	0.00017	5.6×10^{-8}
Explosion (FB-Line)	3.3×10^{-10}	2.7×10^{-6}	7.6×10^{-10}
Design-basis earthquake (F-Canyon)	7.2×10^{-8}	0.00014	6.2×10^{-7}
Design-basis explosion (FB-Line)	3.8×10^{-9}	7.8×10^{-6}	4.7×10^{-8}
Criticality	5.5×10^{-10}	3.0×10^{-6}	1.5×10^{-9}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at SRS, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 7.2×10^{-8} per year (or one chance in 13.9 million that the individual would develop a fatal cancer per year of operation) and 6.2×10^{-7} per year (or one chance in 1.6 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00017 per year (or one chance in 5,880 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The hazardous chemical impacts of potential facility accidents at ANL-W associated with the treatment of driver spent nuclear fuel using electrometallurgical treatment are summarized in **Table 4–35**.

Table 4–35 Hazardous Chemical Impacts Due to Accidents at ANL-W Under Alternative 3

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.008	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1 Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1 Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

The SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) analyzed the consequences of three chemical spills involving hazardous chemicals in the F-Area: (1) the loss of 50 percent sodium hydroxide containment from a skid-mounted 1,000-gallon dumpster; (2) the loss of 50 percent nitric acid containment from a skid-mounted 1,000-gallon dumpster; and (3) the loss of 30 percent sodium nitrite containment from a skid-mounted 1,000-gallon dumpster and an adjacent 1,600-gallon holdup tank. These analyses are summarized in the **Table 4–36**, and are considered representative of wet storage accidents at SRS.

Table 4–36 Hazardous Chemical Impacts Due to Accidents at SRS Under Alternative 3

Accident	*Frequency (Event Per Year)*	*Receptor*	*Exposure*
Wet storage, container rupture	0.005	Noninvolved worker	Sodium hydroxide: less than Permissible Exposure Limit-Time Weighted Average
Wet storage, container rupture	0.005	Noninvolved worker	Nitric acid: less than Permissible Exposure Limit-Time Weighted Average
		Maximally exposed offsite individual	Nitric acid: less than Permissible Exposure Limit-Time Weighted Average
Wet storage, container rupture	0.006	Noninvolved worker	Sodium nitrite: less than Permissible Exposure Limit-Time Weighted Average

Permissible Exposure Limit-Time Weighted Average is used for chemicals having no ERPG values. It is considered to be less than ERPG-1.

Source: DOE 2000.

4.5.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.3.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical treatment and decladding and cleaning processing facilities at ANL-W and the PUREX processing facility at SRS to

be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would not result in disproportionately high and adverse consequences to any particular group within the general population, including minority or low-income populations.

4.5.6 Waste Management

ANL-W

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel assembly hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types generated at ANL-W and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 3 at ANL-W are provided in **Table 4–37**.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–37 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of Alternative 3 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL, prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

Table 4–37 Amounts of Waste Generated at ANL-W Under Alternative 3[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	37.5	13,100
High-level radioactive ceramic waste	16.3 (26 canisters)[b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters)[b]	2,500
Spent nuclear fuel	0	0
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister)[b]	220
Low-level radioactive waste[c]	555	113,000
Transuranic waste	9.1	3,800
Mixed waste	27.5	14,800
Sanitary waste	4,960	1.72×10^{6}
Deactivation Waste		
Low-level radioactive waste[c]	178	60,000
Transuranic waste	1.6	853
Mixed waste	5.1	3,400

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

The waste values in Table 4–37 are total quantities that would be produced as a result of Alternative 3 operations at ANL-W. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 3, the driver spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) at ANL-W for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values. The blanket spent nuclear fuel would be cleaned and declad and sent to SRS for PUREX processing. The high-level radioactive waste that would be generated from PUREX processing at SRS is presented in Table 4–39.

Direct Process Waste

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures. In addition, the blanket spent nuclear fuel cladding is included in the fuel hardware stream.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–37 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste that would be generated as a result of electrometallurgical treatment of driver spent nuclear fuel under this alternative and the driver and blanket spent nuclear fuel generated during the demonstration project at ANL-W would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste could be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste that would be generated is expected to be less than the amount needed to fill a single high-level radioactive waste canister. Conservatively the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of decladding and cleaning blanket spent nuclear fuel and treatment of driver spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from decladding and cleaning blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.5 percent of the total Radioactive Waste Management Complex disposal inventory.

Other Associated Process Transuranic Waste

Transuranic waste would be generated by Alternative 3 from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant facility. The transuranic waste that would be generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is 9.1 cubic meters (321 cubic feet), which is less than 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with decladding and cleaning blanket spent nuclear fuel and the treatment of driver spent nuclear fuel at ANL-W. This would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume is generated over two years. The total deactivation waste would represent an additional 30 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

SRS

The PUREX process at SRS would generate process waste from treatment operations and other associated process waste from support operations. Process waste would include high-level radioactive waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. The associated process waste includes low-level radioactive waste, transuranic waste, and mixed waste. All of the waste streams would be categorized according to existing DOE Orders and SRS waste management procedures. The anticipated categorization of the waste types and their expected interim storage and final disposal locations are given in **Table 4–38**.

Table 4–38 Waste Material Categories at SRS and Interim and Final Locations

Waste Stream	*Category*	*Interim Storage Location*	*Final Disposal Location*
Process Waste			
Liquid waste form	High-level radioactive waste	Initial storage in the high-level radioactive waste Tank Farm followed by post-process storage at the Defense Waste Processing Facility.	Geologic repository
Other Associated Process Waste			
Less than 100 nanocuries per gram transuranic waste	Low-level radioactive waste	None	Low-activity waste vaults
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Transuranic waste storage pads	Waste Isolation Pilot Plant
Contaminated	Mixed waste	Mixed waste storage buildings	Offsite

Estimates of the amounts of waste that would be generated as a result of the PUREX processing at SRS are provided in **Table 4–39**. These values are based on an evaluation of waste forecasts that account only for the fraction of total waste that would be attributable to processing the blanket spent nuclear fuel pins.

As indicated in the following waste type discussions, the amounts of waste associated with this alternative are relatively small compared to onsite and offsite management capacities.

Direct Process Waste

During the PUREX process, liquid high-level radioactive waste would be produced (along with plutonium metal and uranium solution). The liquid waste would be processed in the Defense Waste Processing Facility to yield vitrified high-level radioactive waste (borosilicate glass) and saltstone. This high-level radioactive waste would be temporarily stored at the Defense Waste Processing Facility pending ultimate disposal in a geologic repository. The saltstone is a cement form low-level radioactive waste that is generated or a by-product of SRS tank farm operations. The saltstone would be disposed of on site in the Z-Area Saltstone Vaults. The volume of this saltstone would be about 0.12 percent of the 1.11 million cubic meters (39.2 million cubic feet) storage capacity of the vaults.

Table 4–39 Amounts of Waste Generated at SRS Under Alternative 3

Waste Stream	*Waste Quantities (cubic meters)* [a]
Direct Process Waste	
Liquid high-level radioactive waste	510
Solid high-level radioactive waste [b, c]	5.6 (9 canisters) [b]
Saltstone [c]	1,290
Other Associated Process Waste	
Low-level radioactive waste	900 [d]
Transuranic waste	90
Mixed waste	6.9

[a] These values are estimated based on the heavy metal mass ratio of similar materials processed at SRS (20 metric tons of heavy metal) and provided in DOE 2000.
[b] Standardized high-level radioactive waste (Defense Waste Processing Facility) canisters.
[c] These waste forms result from processing the liquid high-level radioactive waste.
[d] Final disposal volume following a volume reduction (a reduction factor of 4 was assumed).

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated during the PUREX process. The volume of low-level radioactive waste resulting from this alternative (after volume reduction) would be about 3 percent of the total 30,500-cubic meter (1.08 million-cubic foot) disposal capacity of the low-activity waste vaults.

Other Associated Process Transuranic Waste

The volume of transuranic waste that would be generated during the PUREX process would be only about 0.05 percent of the current 168,500-cubic meter (5.95 million-cubic foot) limit for the Waste Isolation Pilot Plant (DOE 1997).

Other Associated Process Mixed Waste

Mixed waste that would be generated during the PUREX process would be temporarily stored on site in the Mixed Waste Storage Buildings prior to eventual offsite disposal. The volume of this waste would be about 0.36 percent of the 1,900-cubic meter (67,100-cubic foot) storage capacity of these storage buildings.

4.6 ALTERNATIVE 4: MELT AND DILUTE BLANKET FUEL AND ELECTROMETALLURGICALLY TREAT DRIVER FUEL AT ANL-W

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be cleaned to remove metallic sodium and then treated using the melt and dilute process at ANL-W. The melt and dilute product from treatment of this fuel would be stored at the Radioactive Scrap and Waste Facility pending repackaging and transportation for disposal in a geologic repository. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel could start in 2003, and subsequent melt and dilute treatment at ANL-W could start in 2005 and could be completed by 2012. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.6.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

Radiological Gaseous Emissions

Cleaning blanket spent nuclear fuel and electrometallurgically treating driver spent nuclear fuel would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The tritium released into the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum annual release of radioactive gaseous emissions would occur when electrometallurgical treatment processing of driver spent nuclear fuel is performed simultaneously with cutting blanket spent nuclear fuel for sodium removal prior to the melt and dilute process. This simultaneous operation would occur over a three-year period during the estimated 10 years of operation starting in 2003. Appendix E, Section E.4.2, provides details on releases during the processing period at ANL-W. Based on an annual blanket spent nuclear fuel processing (e.g., chopping and cleaning) throughput of 10 metric tons of heavy metal and electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 could be released annually to the atmosphere (see Appendix E, Section E.4.2). The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.6.4.

4.6.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with electrometallurgical and melt and dilute treatment processes. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W (see Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to surface water.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. Water consumption for the electrometallurgical and melt and dilute treatment process operations would not impact the current water usage at ANL-W. The current water usage at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to groundwater.

4.6.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

4.6.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–40 and 4–41 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–42 and 4–43. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–44. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.6.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel cleaning and driver spent nuclear fuel chopping and electrorefining. All of these activities are performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when both blanket and driver spent nuclear fuel would be treated simultaneously under this alternative. Appendix E, Section E.4.2, provides details on the treatment process duration and throughputs for each fuel type.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–40**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers

(50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public are releases of tritium gas (about 1 percent of which was conservatively assumed to be in oxidized form) and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–40 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 4

Receptor	*Electrometallurgically Treat Driver Spent Nuclear Fuel*	*Clean and Melt and Dilute Blanket Spent Nuclear Fuel*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem per year) [a]	0.0027	0.00028	0.0030
Excess latent cancer fatalities (per year)	1.4×10^{-6}	1.4×10^{-7}	1.5×10^{-6}
Project total excess latent cancer fatalities [b]	8.0×10^{-6}	3.4×10^{-7}	8.3×10^{-6}
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.00033	0.000048	0.00038
Percent of annual background radiation [c]	0.000092	0.000013	0.00011
Latent cancer fatality risk (per year)	1.7×10^{-10}	2.4×10^{-11}	1.9×10^{-10}
Project total lifetime cancer fatality risk [b]	9.4×10^{-10}	5.8×10^{-11}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.000011	1.2×10^{-6}	0.000012
Latent cancer fatality risk (per year)	5.6×10^{-12}	5.8×10^{-13}	6.2×10^{-12}
Project total lifetime cancer fatality risk [b]	3.3×10^{-11}	1.4×10^{-12}	3.5×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over 13 years. Majority of the dose is a result of 9 years of operation, electrometallurgically treating driver spent nuclear fuel, and cleaning of blanket spent nuclear fuel. Limited offsite consequences are associated with the melt and dilute processing of cleaned spent nuclear fuel.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–41 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with sodium-bonded spent nuclear fuel electrometallurgical and melt and dilute treatment processes. It was concluded that the average worker dose would not be different from what currently is being experienced. The estimated annual collective worker dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 13 years of treatment activities, and the dose (33 person-rem) from one year of deactivation activities was incorporated, the project total worker population dose would be 319 person-rem, leading to a risk of 0.13 latent cancer fatalities.

Table 4–41 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 4

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over 13 years)	0.00031
Worker Population	
Collective dose (person-rem per year) [b]	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	319
Project excess total latent cancer fatalities	0.13

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to an involved worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

[b] The worker dose during one year of facility deactivation would be 33 person-rem.

Source: ANL 1999.

As shown in Tables 4–40 and 4–41:

- The annual dose to the maximally exposed offsite individual would be 0.00038 millirem per year, with an associated risk of developing a fatal cancer of 1.9×10^{-10} per year (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 670,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 319 person-rem, with an associated 0.13 latent cancer fatalities (or one chance in eight that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 4 would be similar to the impacts evaluated for Alternative 1, as described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

4.6.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities associated with cleaning (sodium removal) blanket spent nuclear fuel for melt and dilute processing and treating driver spent nuclear fuel using electrometallurgical treatment are summarized and presented in this section. The detailed analysis of facility accidents, with their associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year and consequence analyses for these two events were not performed. Processing of blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of driver spent nuclear fuel would be performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event, and releases from both the Hot Fuel Examination Facility and the Fuel Conditioning Facility from the single earthquake event. The melt and dilute processing of blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. Melt and dilute processing of the fuel could result in a greater number of accidents to be considered (waste processing-related events) in the assessment of accidents involving blanket spent nuclear fuel at ANL-W than declad and clean operations. The multifacility impacts of the beyond-design-basis earthquake are not relevant to the blanket spent nuclear fuel melt and dilute processing which occurs in only one facility. Therefore, only the higher frequency design-basis earthquake event was analyzed. **Table 4–42** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–43**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 1.9×10^{-6} per year (or one chance in 526,300 that the individual would develop a fatal cancer per year of operation) and 4.9×10^{-8} per year (or one chance in 20.4 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00022 per year (or one chance in 4,545 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The nonradiological impacts of potential facility accidents (hazardous chemical) associated with the treatment of driver spent nuclear fuel using the electrometallurgical treatment process are summarized in **Table 4–44**.

Table 4–42 Accident Frequency and Consequences Under Alternative 4

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}
Design-basis earthquake	0.008	471	0.00024	56.1	0.028	15.2	6.1×10^{-6}
Waste handling spill	0.0024	15	7.5×10^{-6}	1.8	0.00090	0.49	2.0×10^{-7}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency of this accident is the frequency of the facility design-basis earthquake initiating a cell fire.

Table 4–43 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 4

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}
Design-basis earthquake	1.9×10^{-6}	0.00022	4.9×10^{-8}
Waste handling spill	1.8×10^{-8}	2.2×10^{-6}	4.8×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Table 4–44 Nonradiological Impacts of Accidents Under Alternative 4

Accident	*Frequency (Event Per Year)*	*Receptor Location*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.00001	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.6.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address the disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.6.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical and melt and dilute treatment processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would not result in disproportionately high and adverse consequences to any particular group within the general population, including minority or low-income populations.

4.6.6 Waste Management

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. The fuel hardware in this alternative is used as additional steel in the melt and dilute process. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of the waste types generated and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 4 are provided in **Table 4–45**.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–45 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 4 could be volume-reduced at the Waste

Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

Table 4–45 Amounts of Waste Generated at ANL-W Under Alternative 4[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	16.3 (26 canisters) [b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters) [b]	2,500
Melt and dilute product	45.6 (114 canisters) [b]	114,000
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister) [b]	220
Low-level radioactive waste [c]	650	132,000
Transuranic waste	11.2	4,730
Mixed waste	32.1	17,300
Sanitary waste	4,960	1.72×10^{6}
Deactivation Waste		
Low-level radioactive waste [c]	195	66,000
Transuranic waste	1.6	853
Mixed waste	5.6	3,600

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.
[b] Standardized canisters.
[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.
Source: ANL 1999.

The waste values in Table 4–45 are the total quantities that would be produced as a result of Alternative 4 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 4, both the driver and blanket spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic, and a melt and dilute product) for disposal in the repository. In this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For this alternative, fuel assembly hardware would be used as part of the required stainless steel to form the material ingot for disposal of the blanket spent nuclear fuel by melting. Its mass is included as part of the spent nuclear fuel disposal.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste

would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–45 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste and the melt and dilute product that would be generated at ANL-W would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing these materials would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste would be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste that would be generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of processing sodium-bonded spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from electrometallurgical and melt and dilute treatment processing of sodium-bonded spent nuclear fuel at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total 650 cubic meters (22,955 cubic feet) represent approximately 0.6 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste that would be generated at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which would be approximately 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste would be 11.2 cubic meters (395 cubic feet), which would be less than 0.006 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with processing at ANL-W. This waste would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste, generated as a result of dismantling and disposal (electrometallurgical treatment and melt and dilute equipment). Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot hydrostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated in a single year. This waste would represent an increase of approximately three times the annual waste generated by the treatment operations of Alternative 4. The total deactivation waste would represent an additional 30 percent over the total associated process waste requiring disposal.

4.7 ALTERNATIVE 5: DECLAD AND CLEAN BLANKET FUEL AND ELECTROMETALLURGICALLY TREAT DRIVER FUEL AT ANL-W; MELT AND DILUTE BLANKET FUEL AT SRS

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be declad and cleaned to remove metallic sodium at ANL-W, packaged in aluminum cans, and shipped to SRS for treatment using the melt and dilute process at Building 105-L. The melt and dilute product from the treatment process would be stored at SRS pending repackaging and transportation for disposal in a geologic repository. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its shipment to SRS could start in 2003 and could be completed by 2009. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

Current planning at SRS has scheduled the melt and dilute process at Building 105-L for other missions until 2035 (DOE 2000). Melt and dilute process of blanket spent nuclear fuel at SRS could start around 2020, if capacity becomes available, and could be completed by 2023.

4.7.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

At SRS, nonradiological air emissions would result from operation of ancillary support facilities for the melt and dilute process at Building 105-L. The concentrations of nonradiological air pollutants attributed to this alternative at SRS are presented in **Table 4–46**. These concentrations are based on information in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) for the melt and dilute processing of similar fuel (see Appendix E, Section E.5.3.2, for more details). The site boundary concentrations are the incremental concentrations that would be generated in this alternative plus the baseline concentrations given in Section 3.3.3.1. Only those air pollutants that are expected to be emitted under this alternative and have ambient air quality standards are presented in the table. Note that SRS has no Prevention of Significant Deterioration increment-consuming sources on site; therefore, a Prevention of Significant Deterioration increment analysis was not performed. SRS is located in an area of attainment for criteria pollutants; therefore, a confirmatory analysis is not required for this alternative. Health effects from hazardous chemicals associated with this alternative are addressed in Section 4.7.4.1.

Radiological Gaseous Emissions

The decladding and cleaning of the blanket spent nuclear fuel and the electrometallurgical treatment of the driver spent nuclear fuel at ANL-W would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium would be the most prevalent radioactive gaseous fission products released to the environment. The tritium released in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also would contain an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that would be present in the argon cell in nanocuries (10^{-9} curies) and would be released to the atmosphere through the facility stack along with krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions would occur when decladding the blanket spent nuclear fuel for packaging and shipment to SRS and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. It was estimated that this simultaneous operation would occur over a three-year period starting in 2003 (see Appendix E, Section E.4.2, for details on releases during the processing period at ANL-W under this alternative). Based on an annual decladding throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel and an electrometallurgical treatment process of about 0.6 metric tons of heavy metal of driver spent nuclear fuel, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere.

Since declad and cleaned fuel would be packaged and sent to SRS, some gaseous fission products would be expected in that fuel. However, it was conservatively assumed that the gaseous fission products in the blanket spent nuclear fuel also would be released to the environment during the melt and dilute process at SRS. The radiological exposures of the public and workers from radioactive emissions are presented in Section 4.7.4.

4.7.2 Water Resources

As stated in Section 4.4.2, the decladding and cleaning of blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel would not discharge any radiological chemical material to the surface or groundwater at the INEEL site. These activities also would not impact the current groundwater usage at the site. For a discussion of impacts on water resources at ANL-W; see Section 4.4.2.

Table 4–46 Nonradiological Air Quality Concentrations at the Site Boundary Under Alternative 5 at SRS for Comparison With Ambient Air Quality Standards

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutants			
Carbon monoxide	8 hours 1 hour	10,000 40,000	0.08 0.51
Nitrogen dioxide	Annual	100	Less than 0.01
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 150 150	Not detectable Not detectable Not available
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.01 0.03 Not detectable
State-regulated Pollutants			
Gaseous fluoride	30 days 7 days 24 hours 12 hours	0.8 1.6 2.9 3.7	Not detectable Not detectable Not detectable Not detectable
Total suspended particulates	Annual	75	Less than 0.01
Hazardous/Toxic Compounds			
1,1,1-trichloroethane	24 hours	9,550	Less than 0.01
Benzene	24 hours	150	Not detectable
Ethanolamine	24 hours	200	Less than 0.01
Ethyl benzene	24 hours	4,350	Not detectable
Ethylene glycol	24 hours	650	Less than 0.01
Formaldehyde	24 hours	15	Less than 0.01
Glycol ethers	24 hours	No standard	Less than 0.01
Hexachloronaphthalene	24 hours	1	Less than 0.01
Hexane	24 hours	900	Less than 0.01
Manganese	24 hours	25	Not detectable
Methyl alcohol	24 hours	1,310	Less than 0.01
Methyl-ethyl-ketone	24 hours	14,750	Less than 0.01
Methyl-isobutyl-ketone	24 hours	2,050	Not detectable
Methylene chloride	24 hours	8,750	Not detectable
Naphthalene	24 hours	1,250	Less than 0.01
Nitric acid	24 hours	125	Not detectable
Phenol	24 hours	190	Not detectable
Phosphorous	24 hours	0.5	Not detectable
Sodium hydroxide	24 hours	50	Not detectable
Toluene	24 hours	2,000	Less than 0.01
Trichloroethane	24 hours	6,750	Not detectable
Vinyl acetate	24 hours	176	Not detectable
Xylene	24 hours	4,350	Less than 0.01

PM_n = Particulate matter less than or equal to *n* microns in diameter.
Source: Bickford et al. 1997.

The impacts on water resources from treating blanket spent nuclear fuel at Building 105-L using the melt and dilute process are described below.

Surface Water

No surface water would be used for the melt and dilute processing of blanket spent nuclear fuel at Building 105-L. Building 105-L is outside the 100-year floodplain, as shown in Figure 3-6.

Nonradiological Liquid Effluent

No nonradiological liquid effluent would be generated by melting and diluting blanket spent nuclear fuel at Building 105-L. Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility and discharged through an existing NPDES-permitted outfall. Since employment would not increase as a result of processing this fuel, the treatment rates through the Central Wastewater Treatment Facility would not be affected and the requirements of the SRS NPDES permit would continue to be met (DOE 2000).

Radiological Liquid Effluent

No radiological liquid effluent would be discharged to surface water from the melt and dilute process at Building 105-L.

Groundwater

Process water would not be required for the melt and dilute process at Building 105-L. Domestic water would come from groundwater. No increase in domestic water use is anticipated since no increase in employment is expected to result from the melt and dilute operation.

Nonradiological Liquid Effluent

No nonradiological chemicals would be discharged to groundwater from the melt and dilute processing at Building 105-L.

Radiological Liquid Effluent

No radiological liquid effluent would be discharged to groundwater from the melt and dilute process at Building 105-L.

4.7.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W and SRS would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around INEEL and SRS.

4.7.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological and chemical impacts from normal operations are presented in Tables 4–47 through 4–48 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–50 through 4–53. The impacts from hazardous chemical releases during accident conditions are similar to those presented in Section 4.5.4.1.

Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.7.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel decladding and cleaning, driver spent nuclear fuel chopping, and electrorefining. All of these activities would be performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public at ANL-W would result when decladding and cleaning of blanket spent nuclear fuel and treatment of driver spent nuclear fuel are performed simultaneously under this alternative. The doses from decladding and cleaning blanket spent nuclear fuel and treating driver spent nuclear fuel at ANL-W would be similar to those presented for Alternative 2 in Section 4.4.4.1.

Calculated maximum annual and projected total radiological impacts to the public are given in **Table 4–47**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W and Building 105-L at SRS in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL or SRS site boundary and receiving the maximum dose). Primary contributors to doses to members of the public at ANL-W would be from releases of tritium gas (about 1 percent of which were assumed conservatively to be in oxidized form) and krypton-85, which together would contribute over 99.9 percent of the total calculated doses. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public would be much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

The blanket spent nuclear fuel would be declad and cleaned at ANL-W, where it is expected that the gaseous fission products would be released. However, for the melt and dilute process, it is assumed conservatively that these gaseous fission products would be released at SRS. The melt and dilute process is assumed to continue for three years. (Appendix E, Section E.4.4, provides the details on the treatment process duration.) To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table.

Table 4–48 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by SRS workers involved with the melt and dilute process. The estimated annual collective worker dose would be 50 person-rem, with an average individual dose of 500 millirem per year for each of the 100 involved workers. If these estimates were projected for maximum process activities over three years, the project total worker population dose would be 150 person-rem, leading to a risk of 0.06 latent cancer fatalities. The estimated annual collective worker dose from decladding and cleaning blanket spent nuclear fuel and electrometallurgically treating driver spent nuclear fuel at ANL-W is 22 person rem, as indicated in Section 4.4.4.1.

As shown in Tables 4–47 and 4–48:

- The annual dose to the maximally exposed offsite individual at ANL-W would be 0.00038 millirem per year, with an associated risk of developing a fatal cancer of 1.9×10^{-10} per year (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

Table 4–47 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 5

Receptor	*Melt and Dilute Blanket Spent Nuclear Fuel at SRS*	*Clean Blanket Spent Nuclear Fuel and Electrometallurgically Treat Driver Spent Nuclear Fuel at ANL-W*
Population Within 80 Kilometers (50 Miles) in the Year 2010		
Collective dose (person-rem per year) [a]	0.0076	0.0030
Excess latent cancer fatalities (per year)	3.8×10^{-6}	1.5×10^{-6}
Project total excess latent cancer fatalities [b]	0.000011	8.3×10^{-6}
Maximally Exposed Offsite Individual		
Dose (millirem per year) [a]	0.00010	0.00038
Percent of annual background radiation [c]	0.000033	0.00011
Latent cancer fatality risk (per year)	5.0×10^{-11}	1.9×10^{-10}
Project total lifetime cancer fatality risk	1.5×10^{-10}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)		
Dose (millirem per year) [d]	0.000011	0.000012
Latent cancer fatality risk (per year)	5.5×10^{-12}	6.2×10^{-12}
Project total lifetime cancer fatality risk [b]	1.6×10^{-11}	3.5×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over nine years at ANL-W and three years at SRS.
[c] The annual natural background radiation level at INEEL and at SRS is about 360 and 300 millirem, respectively, for the average individual (see Tables 3–8 and 3–20); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem at INEEL and 254,000 person-rem at SRS.
[d] Obtained by dividing the population dose by the number of people projected to live in the year 2010 within 80 kilometers (50 miles) of ANL-W (240,338) and SRS Building 105-L (715,000).

Table 4–48 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 5

Receptor	*Impacts*	
Worker [a]	Operations at SRS	Operations at ANL-W
Average worker dose (millirem per year)	500	60
Average worker latent cancer fatality risk (project total)	0.00060 [b]	0.00022 [c]
Worker Population		
Collective dose (person-rem per year)	50	22
Excess latent cancer fatalities (per year)	0.020	0.0088
Project total dose (person-rem)	150	231
Project total excess latent cancer fatalities	0.06 [b]	0.092

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.
[b] Operations at SRS to treat blanket spent nuclear fuel using melt and dilute processing at Building 105-L would be performed over three years.
[c] Operations at ANL-W to declad and clean blanket spent nuclear fuel and treat driver spent nuclear fuel would be performed over nine years. The project total dose includes 33 person-rem from one year of facility deactivation activities.

Sources: ANL 1999, DOE 2000.

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 670,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

- The collective dose to ANL-W facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to ANL-W facility workers would be 231 person-rem, with an associated 0.092 latent cancer fatalities (or one chance in 11 that the exposed workers would experience a fatal cancer).

- The annual dose to the maximally exposed offsite individual from melt and dilute processing at Building 105-L would be 0.00010 millirem per year, with an associated risk of developing a fatal cancer of 5×10^{-11} per year (or one chance in 20 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of Building 105-L would be 0.0076 person-rem per year, with an associated 3.8×10^{-6} latent cancer fatalities per year (or one chance in 263,100 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of Building 105-L would be 0.000011 latent cancer fatalities (or one chance in 91,000 that the exposed population would experience a fatal cancer).

- The collective dose to Building 105-L facility workers would be 50 person-rem per year, with an associated 0.020 latent cancer fatalities (or one chance in 50 that the workers would experience a fatal cancer per year of operation).

- The project total dose to Building 105-L facility workers would be 150 person-rem, with an associated 0.06 latent cancer fatalities (or one chance in 17 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 5 would be similar to the impacts evaluated for Alternative 1, as described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing baseline chemical environment is presented in Section 3.2.10.2.

For SRS, both carcinogenic and noncarcinogenic health effects to the public were assessed from exposure to hazardous chemicals, and the results are summarized in **Table 4–49**. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released at SRS

under this alternative (see Appendix E, Section E.5.3, for more details). The existing baseline chemical environment is presented in Section 3.3.10.2.

Table 4–49 Hazardous Chemical Impacts to the Public From Operational Activities at SRS Under Alternative 5

Chemical	*Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
Formaldehyde	1.3×10^{-6}	None	1.6×10^{-8}
Hexane	1.3×10^{-6}	6.3×10^{-6}	None
Manganese	Not detectable	Not detectable	None
Methyl ethyl ketone	1.3×10^{-6}	1.3×10^{-6}	None
Naphthalene	1.3×10^{-6}	0.00042	None
Toluene	1.3×10^{-6}	3.1×10^{-6}	None
Hazard Index		0.00043	Not applicable

4.7.4.2 Facility Accidents

Radiological Impacts

The potential radiological impacts to the public and a noninvolved onsite worker resulting from accidents during decladding and cleaning and melting and diluting the blanket spent nuclear fuel elements, and from electrometallurgical treatment of driver spent nuclear fuel operational activities at ANL-W and SRS, are summarized and presented in this section. The detailed analysis of facility accidents, with associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for these two events were not performed. Processing of the blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of the driver spent nuclear fuel is performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because the processing of the driver spent nuclear fuel would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multi-facility impacts of this event. The decladding and cleaning of the blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. The multi-facility impacts of the beyond-design-basis earthquake are not relevant to this blanket spent nuclear fuel processing. Therefore, only the higher frequency design-basis earthquake event was analyzed. **Table 4–50** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 foot) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–51**.

Table 4–50 Accident Frequency and Consequences at ANL-W Under Alternative 5

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this accident is the frequency for the facility design-basis earthquake-initiating cell fire.

Table 4–51 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 5

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would experience a fatal cancer per year of operation).

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during melt and dilute operational activities at SRS are summarized below. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. **Table 4–52** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 300 meters (980 feet) from the facility. The 300-meter (980-foot) distance leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

Table 4–52 Accident Frequency and Consequences at SRS Under Alternative 5

Accident	*Frequency (event per year)*	*Maximally Exposed Offsite Individual*		*Population Within 80 Kilometers (50 Miles)*		*Noninvolved Worker*	
		Dose (millirem)	*Latent Cancer Fatality Risk* [a]	*Dose (person-rem)*	*Excess Latent Cancer Fatalities* [b]	*Dose (millirem)*	*Latent Cancer Fatality Risk* [a]
Waste handling spill	0.0064	2.1	1.1×10^{-6}	3.6	0.0018	0.17	6.8×10^{-8}
Loss of power	0.006	2100	0.0011	3500	1.8	140	0.000056
Melter eruption/explosion [c]	0.0005	269	0.00014	1160	0.58	72.9	0.000029
Fire	0.075	86	0.000043	140	0.07	6.3	2.5×10^{-6}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] In the draft EIS, this accident was identified as "loss of cooling water." Consistent with the SRS Spent Nuclear Fuel Management Final EIS, the accident name was changed.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the facility worker population dose due to accidents. The accident risks for the same receptors are summarized in **Table 4–53**.

Table 4–53 Annual Cancer Risks Due to Accidents at SRS Under Alternative 5

Accident	*Maximally Exposed Offsite Individual* [a]	*Population Within 80 Kilometers (50 Miles)* [b]	*Noninvolved Worker* [a]
Waste handling spill	6.7×10^{-9}	0.000012	5.5×10^{-10}
Loss of power	6.6×10^{-6}	0.011	3.4×10^{-7}
Melter eruption/explosion	7.0×10^{-8}	0.00029	1.5×10^{-8}
Fire	3.2×10^{-6}	0.0053	1.9×10^{-7}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at SRS, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 6.6×10^{-6} per year (or one chance in 151,500 that the individual would develop a fatal cancer per year of operation) and 3.4×10^{-7} per year (or one chance in 2.9 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.011 per year (or one chance in 91 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

The impacts of accidents involving hazardous chemicals for this alternative are the same as those described in Section 4.5.4.2 for Alternative 3: Declad and clean blanket fuel and electrometallurgically treat driver fuel at ANL-W; PUREX process blanket fuel at SRS.

4.7.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.7.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical treatment and decladding and cleaning processing facilities at ANL-W and of the melt and dilute treatment facility at SRS to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.7.6 Waste Management

ANL-W

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types generated at ANL-W and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 5 at ANL-W are provided in **Table 4–54**.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–54 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 5 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

The waste values in Table 4–54 are total quantities that would be produced as a result of Alternative 5 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 5, the driver spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) at ANL-W for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values. The blanket spent nuclear fuel would be cleaned and declad and sent to SRS for melt and dilute processing. The high-level radioactive waste (melt and dilute product) that would be generated from melt and dilute processing at SRS is presented in Table 4–56.

Table 4–54 Amounts of Waste Generated at ANL-W Under Alternative 5[a]

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	37.5	13,100
High-level radioactive ceramic waste	16.3 (26 canisters)[b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters)[b]	2,500
Spent nuclear fuel	0	0
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister)[b]	220
Low-level radioactive waste[c]	555	113,000
Transuranic waste	9.1	3,800
Mixed waste	27.5	14,800
Sanitary waste	4,960	1.7×10^6
Deactivation Waste		
Low-level radioactive waste[c]	178	60,000
Transuranic waste	1.6	853
Mixed waste	5.1	3,400

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

Direct Process Waste

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These hardware components would be primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures. In addition, the blanket spent nuclear fuel cladding would be included in the fuel hardware stream.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both of these waste types would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–54 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel under this alternative, and the driver and blanket spent nuclear fuel from the demonstration project at ANL-W, would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination

of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste could be generated as a result of blanket spent nuclear fuel processing at ANL-W and SRS. This would result from activities in the Hot Fuel Examination Facility (at ANL-W) and Building 105-L (at SRS). Material in this waste stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste that would be generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of decladding and cleaning blanket spent nuclear fuel and electrometallurgically treating driver spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from decladding and cleaning of blanket spent nuclear fuel and electrometallurgical treating driver spent nuclear fuel at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters per year (1,766 cubic feet per year). This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.7 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated at ANL-W would be packaged and certified in accordance with the Waste Isolation Pilot Plant acceptance criteria prior to transport to the Waste Isolation Pilot Plant facility. The transuranic waste generated would amount to approximately 1 cubic meter per year (35 cubic feet per year), which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste would be approximately 9.1 cubic meters (321 cubic feet), which is approximately 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would

be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities at ANL-W. This would include process equipment and process material such as electrorefiner salt and cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot hydrostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Deactivation of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management.

The deactivation waste volume would be generated over a period of two years. The total deactivation waste would represent an additional 30 percent over the total associated process waste requiring disposal.

SRS

The melt and dilute process at SRS would generate process waste from treatment operations and other associated process waste from support operations. Process waste would include metallic high-level radioactive waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. The associated process waste would include low-level radioactive waste, transuranic waste, and mixed waste. All of the waste streams would be categorized according to existing DOE Orders and SRS waste management procedures. The anticipated categorization of the waste types and their expected interim storage and final disposal locations are given in **Table 4–55**.

Table 4–55 Waste Material Categories at SRS and Interim and Final Locations

Waste Stream	*Category*	*Interim Storage Location*	*Final Disposal Location*
Process Waste			
Melt and dilute product	High-level radioactive metallic waste	L-Area	Geologic repository
Off-gas filters	High-level radioactive waste [a]	L-Area	Geologic repository
Other Associated Process Waste			
Less than 100 nanocuries per gram transuranic waste Contaminated	Low-level radioactive waste	None	Low-activity waste vaults
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Transuranic waste storage pads	Waste Isolation Pilot Plant
Offsite	Mixed waste	Mixed waste storage buildings	Offsite

[a] Cleaning of the contaminated filters would generate high-level radioactive liquid waste.

Estimates of the amounts of waste generated as a result of the melt and dilute processing at SRS are provided in **Table 4–56**. These values are based on an evaluation of waste forecasts that accounts only for the fraction of total waste that would be attributable to processing the blanket spent nuclear fuel pins.

Table 4–56 Amounts of Waste Generated at SRS Under Alternative 5

Waste Stream	*Waste Quantities (cubic meters)* [a]
Direct Process Waste	
Melt and dilute product	76 (189 canisters) [b]
Liquid high-level radioactive waste	30 [c]
Saltstone [d]	78
Other Associated Process Waste	
Low-level radioactive waste	330 [e]
Transuranic waste	16.5
Mixed waste	3

[a] Except for the number of canisters of melt and dilute products, the values given are estimated based on the heavy metal mass ratio of similar material processed at SRS (20 metric tons of heavy metal) and provided in DOE 2000.
[b] Standardized spent nuclear fuel canisters.
[c] This is a liquid high-level radioactive waste volume which results in about one Defense Waste Processing Facility borosilicate glass high-level radioactive waste canister or a solid high-level radioactive waste volume of 0.62 cubic meters.
[d] This is a secondary process waste from processing the high-level radioactive waste.
[e] Final disposal volume following a volume reduction (a reduction factor of 4 was assumed).

As indicated in the following waste-type discussions, the amounts of waste associated with this processing alternative are relatively small compared to onsite and offsite management capacities.

Direct Process Waste

During the melt and dilute process, a high-level radioactive waste melt and dilute product (metallic waste) would be the primary product. This waste would be temporarily stored in L-Area prior to ultimate disposition in an offsite (proposed geologic) repository. In addition, some high-level radioactive waste would be generated from cleaning the off-gas filter system, which contains cesium, tellurium, and other isotopes volatilized during the melt and dilute process. The high-level radioactive waste would be processed in the Defense Waste Processing Facility to yield vitrified high-level radioactive waste and saltstone. The vitrified high-level radioactive waste would be temporarily stored at the Defense Waste Processing Facility pending ultimate disposal in a geologic repository. The saltstone is a cement form of low-level radioactive waste that is generated as a by-product of SRS high-level radioactive waste tank form operations. The saltstone would be disposed of on site in the Z-Area Saltstone Vaults. The volume of this saltstone would be about 0.0070 percent of the 1.11 million-cubic meter (39.2 million-cubic foot) disposal capacity of the low-activity waste vaults.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated during the melt and dilute process. The volume of low-level radioactive waste resulting from this alternative (after volume reduction) would be about 1.1 percent of the total 30,500-cubic meter (1.08 million-cubic foot) disposal capacity of the low-activity waste vaults.

Other Associated Process Transuranic Waste

The volume of transuranic waste that would be generated during the melt and dilute process would be about 0.01 percent of the current 168,500-cubic meter (5.95 million-cubic foot) limit for the Waste Isolation Pilot Plant (DOE 1997).

Other Associated Process Mixed Waste

Mixed waste that would be generated during the melt and dilute process would be temporarily stored on site in the Mixed Waste Storage Buildings prior to eventual offsite disposal. The volume of this waste would be about 0.16 percent of the 1,900-cubic meter (67,100-cubic foot) storage capacity of these storage buildings.

4.8 ALTERNATIVE 6: MELT AND DILUTE BLANKET AND DRIVER FUEL AT ANL-W

Under this alternative, sodium-bonded blanket and driver spent nuclear fuel would be treated using the melt and dilute process at ANL-W. The melt and dilute product generated by this treatment process would be stored at the Radioactive Scrap and Waste Facility pending repackaging and transportation for disposal in a geologic repository. Both blanket and driver spent nuclear fuel would be cleaned to remove metallic sodium to the extent possible. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The preparation of driver and blanket spent nuclear fuel to remove metallic sodium could start in 2003. The treatment of driver and blanket spent nuclear fuel by melt and dilute processing at ANL-W could start in 2005 and could be completed by 2015. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.8.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, with the exception that there would be no cadmium release, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

Radiological Gaseous Emissions

The cleaning of the blanket and driver spent nuclear fuel and the melt and dilute treatment of this fuel would release gaseous fission products to the hot argon cell environment. Krypton-85 and elemental tritium would be the most prevalent radioactive gaseous fission products released to the environment. The tritium released into the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also would contain an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that would be present in the argon cell in nanocuries (10^{-9} curies) and would be released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum release of radioactive gases would occur when cutting of blanket and driver spent nuclear fuel to remove metallic sodium is performed simultaneously. This simultaneous operation could occur over a 2-year period during the estimated 10 years of operation, starting in 2003. Appendix E, Section E.4.5, provides more details on various releases during the processing period at ANL-W. Based on an annual processing throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and about 1.7 metric tons of heavy metal of driver spent nuclear fuel elements, about 2,162 curies of elemental tritium and 32,650 curies of gaseous krypton-85 would be released annually to the atmosphere (see Appendix E, Section E.4.5).

4.8.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with the melt and dilute treatment process. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W (see also Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous materials from being released to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent generated by melt and dilute treatment process operations would be discharged to surface water at ANL-W.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. Water consumption for the melt and dilute treatment process operations would not impact the current water usage at ANL-W. The current water usage at ANL-W is 188 million liters per year (49.6 million gallons per year).

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by melt and dilute treatment process operations would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent generated by melt and dilute treatment process operations would be discharged to groundwater.

4.8.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around INEEL.

4.8.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–57 and 4–58 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–59 and 4–60. The impacts from hazardous

chemical releases during accident conditions are presented in Table 4–61. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.8.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel and driver spent nuclear fuel cleaning and melt and dilute processes. All of these activities would be performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when both blanket and driver spent nuclear fuel are treated simultaneously under this alternative. Appendix E, Section E.4.5, provides the details on treatment process duration and throughputs for each fuel type.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–57**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public would be from releases of tritium gases (about 1 percent of which were assumed conservatively to be in oxidized form) and krypton-85; together they contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–58 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by the ANL-W workers involved with the melt and dilute treatment processes. It was concluded that the average worker dose would not be different from that currently being experienced. The estimated annual collective worker dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 12 years of treatment activities (assuming operations would start in 2003 and end in 2015) and the 33 person-rem from 1 year of deactivation activities were included, the project total worker population dose would be 297 person-rem, leading to a risk of 0.12 latent cancer fatalities.

As shown in Tables 4–57 and 4–58:

- The annual dose to the maximally exposed offsite individual would be 0.0020 millirem per year, with an associated risk of developing a fatal cancer of 1×10^{-9} per year (or one chance in 1 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.012 person-rem per year, with an associated 6×10^{-6} latent cancer fatalities per year (or one chance in 167,000 that the population would develop a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.000012 latent cancer fatalities (or one chance in 83,000 that the exposed population would develop a fatal cancer).

Table 4–57 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 6

Receptor	*Melt and Dilute Driver Spent Nuclear Fuel at ANL-W*	*Melt and Dilute Blanket Spent Nuclear Fuel at ANL-W*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem) [a]	0.012	0.00028	0.012
Excess latent cancer fatalities (per year)	6.0×10^{-6}	1.4×10^{-7}	6.1×10^{-6}
Project total excess latent cancer fatalities [b]	0.000012	3.4×10^{-7}	0.000012
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.002	0.000048	0.0020
Percent of annual background radiation [c]	0.00056	0.000013	0.00057
Latent cancer fatality risk (per year)	1.0×10^{-9}	2.4×10^{-11}	1.0×10^{-9}
Project total lifetime cancer fatality risk	2.0×10^{-9}	5.8×10^{-11}	2.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.00005	1.2×10^{-6}	0.000051
Latent cancer fatality risk (per year)	2.5×10^{-11}	5.8×10^{-13}	2.6×10^{-11}
Project total lifetime cancer fatality risk [b]	5.0×10^{-11}	1.4×10^{-12}	5.1×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated dose over 12 years. Nearly all of the impacts are associated with releases of tritium, krypton, and iodine that would occur during the cleaning process. The impact of releases resulting from melt and dilute processing only are not significant.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–58 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 6

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over 12 years)	0.00029
Worker Population	
Collective dose (person-rem per year) [b]	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	297
Project total excess latent cancer fatalities	0.12

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.
[b] Increases to 33 person-rem for one year of deactivation activities.

Source: ANL 1999.

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 297 person-rem with an associated 0.12 latent cancer fatalities (or one chance in eight that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 6 would be similar to the impacts evaluated for Alternative 1, with the exception that there would be no cadmium release, as described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

4.8.4.2 Facility Accidents

Radiological Impacts

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities related to melt and dilute processing of fuel elements are summarized and presented in this section. The detailed analysis of facility accidents and the associated assumptions are presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash event was determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for this event were not performed. Double-batching of the driver spent nuclear fuel was determined potentially to result in a criticality event (see Appendix F), and this event was analyzed for the driver spent nuclear fuel only. Processing of the blanket and driver spent nuclear fuel would be performed in the Hot Fuel Examination Facility. The multifacility impacts of the beyond-design-basis earthquake are not relevant to processing of the fuel under this option. Therefore, only the higher frequency design-basis earthquake event was analyzed. **Table 4–59** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents. (See the discussion on the involved worker in Section 4.2.4.2.) The accident risks for the same receptors are summarized in **Table 4–60**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and a noninvolved worker would be 0.000076 per year (or one chance in 13,160 that the individual would develop a fatal cancer per year of operation) and 2.7×10^{-6} per year (or one chance in 370,400 that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0090 per year (or one chance in 111 that the population would experience a fatal cancer per year of operation).

Table 4–59 Accident Frequency and Consequences Under Alternative 6

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Waste handling spill	0.0024	597	0.00030	70.8	0.035	26.7	0.000011
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	19000	0.0095	2250	1.1	840	0.00034
Sodium fire[c]	0.008	282	0.00014	33	0.016	2.6	1.0×10^{-6}
Criticality	0.0003	0.52	2.6×10^{-7}	0.085	0.000043	0.47	1.9×10^{-7}
Blanket Spent Nuclear Fuel							
Waste handling spill	0.0024	15	7.5×10^{-6}	1.8	0.00090	0.49	2.0×10^{-7}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Design-basis earthquake	0.008	471	0.00024	56.1	0.028	15.2	6.1×10^{-6}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this event is the frequency for the facility design-basis earthquake-initiating cell fire.

Table 4–60 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 6

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Waste liquid spill	7.2×10^{-7}	0.000085	2.6×10^{-8}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	0.000076	0.0090	2.7×10^{-6}
Sodium fire	1.1×10^{-6}	0.00013	8.3×10^{-9}
Criticality	8.0×10^{-11}	1.3×10^{-8}	5.7×10^{-11}
Blanket Spent Nuclear Fuel			
Waste liquid spill	1.8×10^{-8}	2.2×10^{-6}	4.8×10^{-10}
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Design-basis earthquake	1.9×10^{-6}	0.00023	4.8×10^{-8}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendixes F, Section F.3.1.2 for details).

The hazardous chemical impacts of potential facility accidents associated with the treatment of the driver spent nuclear fuel using the electrometallurgical process are summarized in **Table 4–61**.

Table 4–61 Hazardous Chemical Impacts Due to Accidents at ANL-W Under Alternative 6

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Sodium fire	0.008	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.8.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.8.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the melt and dilute treatment processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.8.6 Waste Management

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include high-level radioactive metallic and ceramic waste from stabilizing the residual waste from the existing Electrometallurgical Demonstration Project. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 6 are provided in **Table 4–62**.

Estimates of the total amount of other associated process waste generated are based on an evaluation of waste forecasts from ANL-W, together with an understanding of melt and dilute process activities resulting in the generation of each waste category. The values in Table 4–62 are for disposal and include volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 6 could be volume-reduced by up to 100 percent at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

Table 4–62 Amounts of Waste Generated at ANL-W Under Alternative 6[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	19.4 (31 canisters) [b]	29,000
High-level radioactive metallic waste	0.6 (1 canister) [b]	460
Melt and dilute product	65.6 (164 canisters) [b]	136,400
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister) [b]	220
Low-level radioactive waste [c]	711	144,000
Transuranic waste	12.5	5,400
Mixed waste	35.3	19,000
Sanitary waste	4,960	1.72×10^{6}
Deactivation Waste		
Low-level radioactive waste [c]	213	72,000
Transuranic waste	1.6	853
Mixed waste	5.9	3,500

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

The waste values in Table 4–62 represent total quantities that would be produced as a result of Alternative 6 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 6, both the driver and blanket spent nuclear fuel would be transformed into a high-level radioactive waste form (melt and dilute product) for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For this alternative, fuel hardware would be used as part of the stainless steel to form the metal ingot for disposal of the fuel by melting. Its mass is included as part of the spent nuclear fuel disposal.

Under this alternative, metallic and ceramic high-level radioactive waste would be produced from existing process material at ANL-W. This waste would be generated to stabilize materials produced during the electrometallurgical demonstration project. In addition, the salt removed from the melting furnace used for driver spent nuclear fuel would contain fission products that would be stabilized in ceramic waste. The volumes of waste forms provided in Table 4–62 are for the standardized canisters required for disposal of these materials.

A second metallic high-level radioactive waste called the melt and dilute product would be generated as a result of the melt and dilute treatment of fuel at ANL-W. This waste, along with the ceramic and metallic waste from the demonstration project, would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a

combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing this high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste would be generated as a result of driver and blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbent used in the off-gas system which would have collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, was used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of the melt and dilute treatment of fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 711 cubic meters (25,100 cubic feet) represents approximately 0.6 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant acceptance criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste would be 12.5 cubic meters (441 cubic feet), which is less than 0.008 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive

Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment are identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with melt and dilute treatment of sodium-bonded spent nuclear fuel at ANL-W. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of dismantling and disposing of electrometallurgical treatment and melt and dilute processing equipment. Components of the electrometallurgical demonstration project that would require disposition include two electrorefiners; two hot hydrostatic presses; and one V-mixer, as well as other components such as the grinder/crusher. Deactivation of components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The total deactivation waste would represent 35 percent over the total associated process waste requiring disposal.

4.9 TRANSPORTATION IMPACTS

Transportation impacts may be divided into two parts: the impacts of incident-free or routine transportation, and the impacts of transportation accidents. Incident-free transportation and transportation accident impacts are divided into two parts: nonradiological impacts and radiological impacts. Incident-free transportation impacts include radiological impacts on the public and the crew from the radiation field that surrounds the package. Nonradiological impacts of incident-free transportation are from vehicular emissions. Nonradiological impacts of potential transportation accidents include traffic accident fatalities. Only in the worst conceivable conditions, which are of low probability, could a transportation cask of the type used to transport radioactive material be so damaged that a release of radioactivity to the environment could occur.

The impact of a specific accident is expressed in terms of probabilistic risk, which is the probability of that accident occurring multiplied by its consequence. Hypothetical accidents ranging from a low-speed impact to those involving high-speed impacts with or without fires leading to cask failure are analyzed. The accident frequencies and consequences are evaluated using the method developed for the NRC, which is known as the "Modal Study" (NRC 1987). The overall risk is obtained by summing the individual risks from all accident conditions. The risks for radiological accidents are expressed as additional latent cancer fatalities and as additional immediate fatalities for nonradiological accidents. The risks of incident-free effects are expressed in additional latent cancer fatalities.

The first step in the ground transportation analysis was to determine the incident-free and accident risk factors on a per-shipment basis for transportation of the various materials. Calculation of risk factors was accomplished by using the HIGHWAY (Johnson et al. 1993) computer codes to choose representative routes according to U.S. Department of Transportation regulations. These codes provide population estimates so that RADTRAN 5 (Neuhauser and Kanipe 1998) codes could be used to determine the radiological risk factors. This analysis is discussed in Appendix G. **Table 4–63** lists the sodium-bonded spent nuclear fuel destinations evaluated in this EIS.

Table 4–63 Transportation Summary for Sodium-Bonded Fuel

Fuel Type	*Alternatives* [a]	*Metric Tons of Heavy Metal*	*Origin/State*	*Destination/ State*	*Cask*	*Number of Shipments/Type of Transport*
EBR-II driver	All	1.1	ANL-W/ID	ANL-W/ID	HFEF-5	84/On site, intrafacility transfers
EBR-II driver	All	2.0	INTEC/ID	ANL-W/ID	TN-FSV or NAC-LWT	17/On site with roads open or 43/On site with roads open
EBR-II blanket	All	22.4	ANL-W/ID	ANL-W/ID	HFEF-5	165/On site, intrafacility transfers
Fast Flux Test Facility driver [b]	All	0.33	Hanford/WA	ANL-W/ID	T-3	10/Public highways
Fermi-1 blanket	All	34.2	INTEC/ID	ANL-W/ID	PB-1	14/On site with road closed
Miscellaneous [b]	All	0.1	Oak Ridge National Laboratory/TN, Sandia National Laboratories/NM, SRS/SC	ANL-W/ID	To be determined by DOE	1/Public highways 1/Public highways 1/Public highways
Declad EBR-II blanket	3 and 5	22.4	ANL-W/ID	SRS/SC	NAC-LWT	11/Public highways
Declad Fermi-1 blanket	3 and 5	34.2	ANL-W/ID	SRS/SC	NAC-LWT	18/Public highways

ID = Idaho; NM = New Mexico; SC = South Carolina; TN = Tennessee; WA = Washington

[a] "All" includes the proposed action plus the No Action Alternative.

[b] This fuel is assumed to be in Idaho per the amended Record of Decision for the Programmatic Spent Nuclear Fuel EIS (61 FR 9441).

Transportation of the Fast Flux Test Facility driver spent nuclear fuel currently stored at the Hanford site and the small amounts of miscellaneous sodium-bonded spent nuclear fuel currently stored at Oak Ridge National Laboratory, Sandia National Laboratories, and at SRS (see Section 2.2.3 and Appendix D, Section D.5.2, for more details on miscellaneous fuel types) are shipment campaigns related to sodium-bonded spent nuclear fuel and were analyzed by DOE in the Programmatic Spent Nuclear Fuel EIS (DOE 1995a and 61 FR 9441), so they are included by reference in this impact analysis. See Appendix G for more details.

All EBR-II blanket and some EBR-II driver spent nuclear fuel are currently stored at ANL-W and would be subject to a building-to-building movement for processing. Since the movement is a short distance on closed DOE-controlled roads, DOE procedures and the NRC regulations do not require the use of a certified Type B cask. No incident-free risk analysis is necessary because the public would receive no measurable exposure. The worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. The probability and consequences of potential accidents during movement are bounded in frequency and consequence by handling accidents.

Fermi-1 blanket spent nuclear fuel would be shipped from INTEC to ANL-W in the Type B cask (PB-1 Cask). Since DOE would close the roads between INTEC and ANL-W using existing traffic gates, and the road is uninhabited, no quantitative analysis is necessary. No incident-free risk analysis is necessary because the public would receive no measurable exposure. The worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. Once the cask is closed

for movement on the closed roads, the likelihood and consequences of any foreseeable accident would be very small and are not further quantified.

EBR-II driver spent nuclear fuel would be shipped from INTEC to ANL-W in a certified Type B cask, either model TN-FSV or model NAC-LWT. Since the cask would be certified, DOE would not close the roads between INTEC and ANL-W. However, since the road is uninhabited, limited quantitative analysis is necessary. No incident-free risk analysis for exposure to the public at stops or in their homes is necessary. The worker dose is analyzed for the transportation crew, and the dose to other vehicles using the road is estimated. No accident analysis is necessary because potential accidents during movement are bounded in frequency and consequence by the handling accidents. Once the cask is closed for movement on the closed roads, the likelihood and consequences of any foreseeable accident would be very small.

Transportation-related risks are calculated and presented separately for workers and members of the general public. The workers considered are truck crew members involved in the actual overland transportation. The general public includes all persons who could be exposed to a shipment while it is moving or stopped during transit. The affected population includes individuals living within 800 meters (0.5 miles) of each side of the road. Potential risks are estimated for the affected population and for the hypothetical maximally exposed offsite individual. For incident-free operations, the maximally exposed individual would be an individual stuck in traffic next to the shipment for 30 minutes. For accident conditions, the maximally exposed individual is assumed to be an individual located 33 meters (105 feet) directly downwind from the accident. The risk to the affected population is a measure of the radiological risk posed to society as a whole by the alternative being considered. The impact to the affected population is used as the primary means of comparing various alternatives.

The following provides a summary of transportation impacts. Appendix G details the methods and assumptions used.

4.9.1 Onsite Transportation Impacts - No Action Alternative

Under all alternatives, EBR-II driver spent nuclear fuel would be shipped by DOE in 17 shipments using the TN-FSU cask, or 43 shipments using the NAC-LWT cask. The analysis assumes that 43 shipments are made. Fifteen ceramic waste form, 1 metallic waste form, and 355 spent nuclear fuel shipments would be made from ANL-W to the INEEL Dry Transfer Facility. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 16,000 kilometers (9,900 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.003 person-rem; the collective dose to the affected population would be 0.022 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.2×10^{-6} latent fatal cancers among transportation workers and 0.000011 latent fatal cancers in the total affected population over the duration of the transportation activities. Latent fatal cancers associated with radiological releases were estimated by multiplying the worker dose by 0.0004 latent fatal cancers per person-rem of exposure, and multiplying the collective dose to the affected population by 0.0005 latent fatal cancers per person-rem of exposure (ICRP 1991).

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00012 traffic fatalities.

4.9.2 Onsite Transportation Impacts - Alternative 1

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 125 ceramic waste form and 5 metallic waste form shipments from ANL-W to the INEEL Dry Transfer Facility would be made. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 6,700 kilometers (4,200 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities entailed by this alternative was estimated at 0.0044 person-rem; the collective dose to the affected population would be 0.033 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.8×10^{-6} latent fatal cancers among transportation workers and 0.000016 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.000052 traffic fatalities.

4.9.3 Onsite Transportation Impacts - Alternative 2

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 27 ceramic waste form, 7 metallic waste form, and 63 spent nuclear fuel shipments from ANL-W to the INEEL Dry Transfer Facility would be made. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 5,200 kilometers (3,200 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities entailed by this alternative was estimated at 0.0043 person-rem; the collective dose to the affected population would be 0.032 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.7×10^{-6} latent fatal cancers among transportation workers and 0.000016 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00008 traffic fatalities.

4.9.4 On- and Offsite Transportation Impacts - Alternative 3

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, Alternative 3 would require 11 shipments of declad EBR-II blanket material and 18 shipments of Fermi-1 blanket material from ANL-W to SRS. Twenty-seven ceramic waste form and 2 metallic waste form shipments from ANL-W to the INEEL Dry Transfer Facility would be made. The impacts for these alternatives include both on- and offsite transportation. The total distance traveled on public roads by trucks carrying radioactive materials would be 111,800 kilometers (69,500 miles).

Impacts of On- and Offsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.0052 person-rem; the collective dose to the affected population would be 0.042 person-rem. Accordingly, incident-free transportation of radioactive material would result in 2.1×10^{-6} latent fatal cancers among transportation workers and 0.000021 latent fatal cancers in the total affected population over the duration of the transportation activities. The dose to transportation workers from transporting cleaned and declad blanket spent nuclear fuel to SRS was estimated at 0.0012 person-rem; the collective dose to the affected population would be 0.012 person-rem. Accordingly, incident-free transportation of radioactive material would result in 4.7×10^{-7} latent fatal cancers among transportation workers and 6×10^{-6} latent fatal cancers in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this alternative is 0.00039.

Impacts of On- and Offsite Accidents During Ground Transportation. The maximum foreseeable offsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would be shipment of EBR-II blanket material from DOE's facility at ANL-W to SRS with a Severity Category 5 accident in a suburban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.00024 person-rem to the public with an associated 1.2×10^{-7} latent fatal cancers, and 2.5×10^{-13} rem to the hypothetical maximally exposed individual with a latent fatal cancer risk of 1.3×10^{-15}. No fatalities would be expected. The probabilities of more severe accidents, different weather conditions at the time of accident, or occurrence in a more densely populated area also were evaluated, and were estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of 3.4×10^{-6} person-rem would result in 1.7×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0018 traffic fatalities. Ground transportation accident risks to the affected population from transporting blanket fuel to SRS were estimated at 3×10^{-6} person-rem, resulting in 1.5×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0017 traffic fatalities.

4.9.5 Onsite Transportation Impacts - Alternative 4

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 27 ceramic waste form, 2 metallic waste form, and 114 melt and dilute product shipments would be made from ANL-W to the INEEL Dry Transfer Facility. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 7,200 kilometers (4,500 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.02 person-rem; the collective dose to the public would be 0.14 person-rem. Accordingly, incident-free transportation of radioactive material would result in 7.9×10^{-6} latent fatal cancers among transportation workers, and 0.000072 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00011 traffic fatalities.

4.9.6 On- and Offsite Transportation Impacts - Alternative 5

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, Alternative 5 requires 11 shipments of declad EBR-II blanket material and 18 shipments of Fermi-1 blanket material from ANL-W to SRS. Twenty-seven ceramic waste form and 2 metallic waste form shipments would be made from ANL-W to the INEEL Dry Transfer Facility. The impacts for these alternatives include both on- and offsite transportation. The total distance traveled on public roads by trucks carrying radioactive materials would be 111,800 kilometers (69,500 miles).

Impacts of On- and Offsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.0052 person-rem; the collective dose to the public would be 0.042 person-rem. Accordingly, incident-free transportation of radioactive material would result in 2.1×10^{-6} latent fatal cancers among transportation workers and 0.000021 latent fatal cancers in the total affected population over the duration of the transportation activities. The dose to transportation workers from transporting cleaned and declad blanket spent nuclear fuel to SRS was estimated at 0.0012 person-rem; the collective dose to the public would be 0.012 person-rem. Accordingly, incident-free transportation of radioactive material would result in 4.7×10^{-7} latent fatal cancers among transportation workers and 6×10^{-6} latent fatal cancers in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this alternative would be 0.00039.

Impacts of On- and Offsite Accidents During Ground Transportation. The maximum foreseeable offsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would be shipment of EBR-II blanket material from DOE's facility at ANL-W to SRS with a Severity Category 5 accident in a suburban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.00024 person-rem to the public with an associated 1.2×10^{-7} latent fatal cancers, and 2.5×10^{-13} rem to the hypothetical maximally exposed individual with a latent fatal cancer risk of 1.3×10^{-15}. No fatalities would be expected to occur. The probabilities of more severe accidents, different weather conditions at the time of accident, or occurrence in a more densely populated area also were evaluated, and were estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of 3.4×10^{-6} person-rem would result in 1.7×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0018 traffic fatalities. Ground transportation accident risks to the affected population from transporting blanket fuel to SRS were estimated at 3×10^{-6} person-rem, resulting in 1.5×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0017 traffic fatalities.

4.9.7 Onsite Transportation Impacts - Alternative 6

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 32 ceramic waste form, 1 metallic waste form, and 164 melt and dilute product shipments would be made from ANL-W to the INEEL Dry Transfer

Facility. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 9,300 kilometers (5,800 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.027 person-rem; the collective dose to the public would be 0.20 person-rem. Accordingly, incident-free transportation of radioactive material would result in 0.000011 latent fatal cancers among transportation workers and 0.0001 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00014 traffic fatalities.

4.10 PREFERRED ALTERNATIVE

DOE has identified electrometallurgical treatment (Alternative 1) as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel; therefore, Fermi-1 blanket spent nuclear fuel would remain in storage, pending a subsequent decision on its long-term management. While EBR-II spent nuclear fuel is undergoing electrometallurgical treatment, DOE has approximately four years[1] in which to evaluate the operating experience of electrometallurgical treatment technology and continue to investigate alternative treatment techniques that currently require additional development for the Fermi-1 blanket spent nuclear fuel. After this data is evaluated, DOE would decide whether to treat the Fermi-1 blanket spent nuclear fuel using electrometallurgical treatment or to use another treatment method and/or disposal technique.

The environmental impacts of the Preferred Alternative, as identified above, are provided in detail in Section 4.2 for the No Action Alternative and in Section 4.3 for Alternative 1. The evaluations provided in these sections cover treatment of both Fermi-1 blanket and other sodium-bonded spent nuclear fuel. The environmental impact contributions from treatment of the Fermi-1 blanket spent nuclear fuel as compared to the EBR-II sodium-bonded spent nuclear are negligible for all resources except for waste management. Overall, the environmental impacts of Alternative 1 bound those of the Preferred Alternative for all resources except for waste management, where the No Action Alternative bounds. The decision to electrometallurgically treat all sodium-bonded spent nuclear fuel except the Fermi-1 blanket fuel would reduce the treatment duration under Alternative 1 from 13 to 7 years. Storing Fermi-1 blanket spent nuclear fuel pending a subsequent decision on its long-term management would not change the duration of the No Action Alternative, i.e., it would remain 35 years.

Should DOE decide to treat Fermi-1 blanket fuel using a treatment method or process, other than electrometallurgical treatment, that was analyzed in Sections 4.4 through 4.7 of this EIS, the environmental consequences would be equal to or bounded by the EIS. As indicated in these sections, all the alternatives analyzed would result in very small and essentially indistinguishable impacts to public and occupational health and safety, air quality, water resources, environmental justice, and transportation. The volumes of waste generated by separate treatment of Fermi-1 blanket fuel would be equal to or bounded by the values

[1] *Even though it would take six years to electrometallurgically treat EBR-II spent nuclear fuel at ANL-W, for planning purposes, DOE would need to make the decision in four years.*

presented for each of the alternatives analyzed in detail. A decision by DOE to treat some or all of the sodium-bonded blanket fuel using a method which has not been analyzed in detail in this EIS would require an evaluation of associated environmental impacts under a separate NEPA document.

4.11 CUMULATIVE IMPACTS

The Council on Environmental Quality regulations implementing NEPA procedural provisions define cumulative impacts as the impacts on the environment which result from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions, regardless of what agency (Federal or nonfederal) or person undertakes such other actions (40 CFR 1508.7). The cumulative impacts analysis presented in this section is based on the incremental contribution from the maximum impacts from the proposed action added to the baseline conditions at ANL-W and SRS, as well as the maximum impacts from other on- and offsite past, present, and other reasonably foreseeable future actions. Although it is unlikely that the alternative with the maximum impacts would be implemented to treat and manage sodium-bonded spent nuclear fuel at ANL-W and SRS, it was used to estimate cumulative impacts to ensure a conservative analysis. In accordance with a handbook recently prepared by the Council on Environmental Quality, DOE identified the resource areas in which the treatment and management of sodium-bonded spent nuclear fuel could add to the impacts of past, present, and reasonably foreseeable actions within the project impact zones, as defined by the Council on Environmental Quality (CEQ 1997).

Based on an examination of the environmental impacts of the proposed action, coupled with DOE and other agency actions, it was determined that cumulative impacts for the following areas need to be presented: (1) air resources, (2) water resources, (3) socioeconomics, (4) public and worker health, (5) environmental justice, and (6) waste generation. Discussions of cumulative impacts for land resources, site infrastructure, geologic resources, ecological resources, and cultural and paleontological resources were omitted because the related impacts from the proposed treatment and management of sodium-bonded spent nuclear fuel would either not occur or be so small that their potential contribution to cumulative impacts would be negligible.

For determining the impacts to air, water, socioeconomic, human health, environmental justice, and waste generation resources from commercial and Federal nuclear facilities, the 80-kilometer (50-mile) radius surrounding ANL-W and SRS was selected as the project impact zone. For liquid releases from SRS, the downstream population that uses the Savannah River as its source of drinking water was included in the SRS project impact zone.

Cumulative transportation impacts are discussed at the end of this section.

4.11.1 ANL-W and INEEL

Significant offsite activities within a 80-kilometer (50-mile) radius of ANL-W and INEEL that potentially would contribute to the cumulative environmental impacts presented in this analysis include the System Integration Corporation quartzite mining operation in Arco Hills and the Food, Machinery, and Chemical Corporation, a phosphate processing operations in Pocatello, Idaho. The Food, Machinery, and Chemical Corporation is a primary source for offsite radiological emissions. These emissions have been evaluated by the EPA. Radiological impacts from the operation of the phosphate processing operations are minimal, and are not included in assessments at INEEL (DOE 1999a).

The counties surrounding ANL-W and INEEL have a number of existing and planned industrial and commercial facilities with permitted air emissions and water usage. Because of the distances between ANL-W and INEEL and the private industrial facilities, there is little opportunity for the interaction of plant emissions and no significant contribution to the cumulative impact on air or water. Reasonably foreseeable offsite actions evaluated in this EIS are presented in **Table 4–64**.

Table 4–64 Offsite Activities Included in the Assessment of Cumulative Impacts at ANL-W and INEEL

Activity	*Description*
Housing development, Idaho Falls	300-unit single family housing development planned on approximately 150 acres of vacant land.
Business park, Rexburg	50 acres of vacant land between two light industrial facilities planned for expansion into a light industrial/business park for 30-40 businesses.
Manufacturer, Pocatello	Existing manufactured-home factory to expand from approximately 50 to between 140 and 150 employees. Expansion of 22 acres in Pocatello Airport Industrial Park.
Food, Machinery, and Chemical Corp., Pocatello	Phosphate manufacturing plant to reduce number of furnaces from four to three within the next two years; 25-30 jobs could be lost.
System Integration Corporation Arco Hills Quartzite Mine	Quartzite mining operation and ore processing near Arco Hills on 56 acres. Fourteen acres would be disturbed by the quarry operation and a small waste ore dump; 22 acres would be disturbed by the construction of a haul road; 11 acres would be disturbed by the ore crushing facilities; and 9 acres would be disturbed by the loading facilities at INEEL. The project would employ 40 workers.

Source: DOE 1999a.

The cumulative impacts analysis also addressed the contributory effects from other past, present, and reasonably foreseeable future DOE actions at ANL-W and INEEL. These actions and their associated NEPA documentation are summarized in Section 1.6. The contributory effects of impacts from actions proposed in the *Idaho High-Level Waste and Facilities Disposition Environmental Impact Statement* (DOE 1999e) were included in the cumulative impact analysis. In this EIS, DOE evaluated the impacts from the proposed construction and operation of a high-level radioactive waste and liquid sodium-bearing waste treatment facility at INEEL to make these materials ready for disposal. This project also involves the disposition of high-level radioactive waste generation, storage, and treatment facilities at INEEL upon the completion of their missions.

Other reasonably foreseeable future actions that may contribute to cumulative impacts at INEEL but were not included in this analysis include a proposed DOE Office of Nuclear Energy, Science, and Technology project. This project involves evaluating INEEL as a potential site for the production of plutonium-238 for use in radioisotope power systems for future space missions. This project would include the use of INEEL's Fluorinel Dissolution Process Facility at INTEC for either storing neptunium-237 and/or fabricating and processing neptunium-237 targets to produce plutonium-238, and the use of the Advanced Test Reactor for the irradiation of neptunium-237 targets. The Advanced Test Reactor is an operating test reactor with a programmatic mission to support the Naval Reactor Fuels Program. Public scoping for this project has been completed. A preliminary review of the project indicates that there would be a contributory effect to the cumulative impacts—primarily to public and worker health and safety due to the fabrication and processing of neptunium-237 targets in the Fluorinel Dissolution Process Facility, loading and unloading targets in the Advanced Test Reactor, and handling of irradiated targets for packaging and shipping. The cumulative impacts from this proposed project will be addressed in a separate NEPA document.

The proposed commercial project (VentureStar) would involve a commercial spin-off of the National Aeronautics and Space Administration's Reusable Launch Vehicle research program that would replace the existing Space Shuttle Program. INEEL is being considered as a potential candidate site for both the launch and landing of this next-generation spacecraft. The project is in the very early stages of development and does not appear to be near term (5 to 10 years). Cumulative impacts from this proposed project would be addressed in separate NEPA documentation.

The cumulative impacts analysis also included the impacts from actions proposed in this EIS. Risks to members of the public and site workers from radiological and nonradiological releases were based on operational impacts from the alternatives described in Chapter 4 of this EIS.

Temporal limits were defined by examining the period of influence from both the proposed action and other Federal and nonfederal actions that have the potential for cumulative impacts. Actions to support the treatment of sodium-bonded blanket spent nuclear fuel at ANL-W are expected to begin in 2000 in preparation for ultimate offsite disposal, possibly in a geologic repository which probably will not be available until at least 2010. Final offsite shipments of spent nuclear fuel at ANL-W and INEEL for disposal would be completed by 2035.

The period of interest for the cumulative impacts analysis for this EIS includes the proposed construction, operation, and disposition of facilities identified in the *Idaho High-Level Waste and Facilities Disposition Environmental Impact Statement* (DOE 1999e) and the *Advanced Mixed Waste Treatment Project Final Environmental Impact Statement* (DOE 1999a), while actions for other nuclear materials and surplus plutonium disposition would be ongoing.

4.11.1.1Air Resources

Table 4–65 compares the cumulative concentrations of nonradiological air pollutants from INEEL to Federal and state regulatory standards. The listed values are the maximum modeled concentrations that could occur at the ground level at the site boundary. The data demonstrate that the total estimated concentrations of nonradiological air pollutants from INEEL in the all cases would be well below the regulatory standards at the site boundary. Among the pollutants, the concentrations of nitrogen oxides come closest to the standard (14 percent of the standard for the annual averaging time). The remaining pollutant emissions would result in concentrations below 13 percent of the applicable standards. As indicated in this table, the values presented in the INEEL baseline include concentrations from releases at ANL-W. ANL-W's criteria pollutant concentrations are from currently operating equipment, which are not expected to increase under any of the alternatives. Therefore, there would be no contribution to cumulative air quality impacts at INEEL as a result of the proposed action.

4.11.1.2Water Resources

There would be no liquid effluent released to surface water or groundwater from the operation of ANL-W or INEEL facilities as a result of the proposed action. Therefore, there would be no contribution to the cumulative impact.

4.11.1.3Socioeconomic Impacts

No additional workers would be required for the operation of ANL-W or INEEL facilities as a result of the proposed action. Therefore, there would be no contribution to the cumulative impact.

4.11.1.4Public and Worker Health

Table 4–66 summarizes the cumulative radiological health effects of routine ANL-W and INEEL operations, proposed DOE actions, and nonfederal nuclear facility operations. Impacts resulting from proposed DOE actions are described in the various EISs listed in Section 1.6. In addition to estimated radiological doses to the maximally exposed offsite individual, the offsite population, and workers, Table 4–66 lists the potential number of latent cancer fatalities for the public and workers from exposure to radiation. The radiation dose to the maximally exposed offsite individual would be 0.047 millirem per year, which is well below the applicable DOE regulatory limits (10 millirem per year from the air pathway [40 CFR 61] and

100 millirem per year for all pathways). The total annual population dose of 0.35 person-rem for current and projected activities translates into 0.00017 latent cancer fatalities for each year of exposure for the population living within a 80-kilometer (50-mile) radius of the ANL-W.

Table 4–65 Estimated Maximum Cumulative Ground-Level Concentrations of Nonradiological Pollutants (micrograms per cubic meter) at the INEEL Boundary

Pollutant	*Averaging Time*	*Most Stringent Standard or Guideline* [a]	*INEEL Baseline* [b]	*Advanced Mixed Waste Treatment Project EIS* [c]	*Idaho High-Level Waste and Facilities Disposition EIS* [d]	*Cumulative Concentrations* [e]
Carbon monoxide	8 hours	10,000 [f]	120	1	4.2	130
	1 hour	40,000 [f]	265	115	10	390
Nitrogen oxides	Annual	100 [f]	13	0.3	0.2	14
PM_{10}	Annual	50 [f]	0.65	0.006	0.02	1
	24 hours	150 [f]	13	4.6	0.3	18
	24 hours (99^{th} percentile over 3 years)	150 [g]	Not available	Not available	Not available	Not available
$PM_{2.5}$	3-year annual	15 [g]	Not available	Not available	Not available	Not available
	24 hours (98^{th} percentile over 3 years)	65 [g]	Not available	Not available	Not available	Not available
Sulfur dioxide	Annual	80 [f]	3.4	0.012	0.57	4
	24 hours	365 [f]	32	4.5	9	46
	3 hours	1,300 [f]	84	25	42	151

PM_n = Particulate matter less than or equal to *n* microns in diameter.

[a] The more stringent Federal or state standard is presented if both exist for the averaging period.

[b] INEEL baseline includes concentrations from releases at ANL-W which, in turn, include releases under all alternatives considered in the SBSNF EIS (see Section 3.2.3).

[c] DOE 1999a: Table 5.7-6, Preferred Alternative (Microencapsulation option).

[d] DOE 1999e: Table C.2-14, Separation (Planning Basis) option.

[e] Values presented in this column could be different from the sum of the individual values due to rounding.

[f] Federal and state standard.

[g] Federal standard.

The annual collective dose to the worker population would be 200 person-rem. In addition, doses to individual workers would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1, which is well below the regulatory limit of 5,000 millirem per year (10 CFR 835). Furthermore, "as low as reasonably achievable" principles would be exercised to maintain individual worker doses below the DOE Administrative Control Level of 2,000 millirem per year.

4.11.1.5Environmental Justice

As discussed in Chapter 4 and Appendix H, implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel at ANL-W and INEEL would have no significant impact on public health or the environment. Therefore, the implementation of the proposed action or the No Action Alternative would result in no disproportionately high and adverse impacts on minority or low-income populations residing within potentially affected areas.

Table 4–66 Estimated Annual Cumulative Radiological Doses and Resulting Health Effects to Offsite Population and Facility Workers at ANL-W and INEEL

	Maximally Exposed Offsite Individual		*Population* [a]		*Workers*	
Activity	*Dose (Millirem)*	*Latent Cancer Fatality Risk*	*Collective Dose (Person-Rem)*	*Excess Latent Cancer Fatalities*	*Collective Dose (Person-Rem)*	*Excess Latent Cancer Fatalities*
ANL-W and INEEL Baseline [b]	0.021	1.1×10^{-8}	0.23	0.00012	115	0.046
SBSNF EIS [c]	0.002	1.0×10^{-9}	0.012	6.0×10^{-6}	22	0.0088
Advanced Mixed Waste Treatment Program [d]	0.022	1.1×10^{-8}	0.009	4.5×10^{-6}	4.1 [e]	0.0016
High-Level Waste and Facilities Disposition [f]	0.002	1.0×10^{-9}	0.10	0.00005	59	0.024
Total	0.047	2.4×10^{-8}	0.35	0.0017	200	0.08

[a] A collective dose to the 80-kilometer (50-mile) population from atmospheric releases. There would be no liquid releases from ANL-W and INEEL facilities as a result of the proposed action.
[b] Data from Tables 3–9 and 3–10 of this SBSNF EIS.
[c] Alternative 6: Melt and dilute blanket and driver fuel at ANL-W.
[d] DOE 1999a: Tables 5.12–1 and E.4–7. Preferred Alternative (Microencapsulation Option) Record of Decision (64 FR 16948).
[e] Estimate based on the number of workers and the average dose per worker, i.e., 50 workers (DOE 1999a: Table E.4–7) × 81 millirem (DOE 1999a: Table 5.12–1) = 4050 person millirem = 4.1 person-rem.
[f] DOE 1999e: Table 5.4–6, maximum dose for any alternative. Average annual dose of 190 millirem per worker.

4.11.1.6Waste Generation

As stated in the Waste Management discussions for each alternative presented earlier in Chapter 4, low-level radioactive waste, mixed and hazardous waste, and transuranic waste would be generated by the treatment of sodium-bonded spent nuclear fuel. Under the proposed action (with the partial exception of Alternative 2), the existing sodium-bonded spent nuclear fuel inventories at ANL-W and INEEL would be converted into a high-level radioactive waste form for disposal in a geologic repository and, therefore, the volume of the high-level radioactive waste that would be generated is not counted as new waste—"high-level radioactive waste." In fact, under the proposed action, the amount of material at ANL-W and INEEL scheduled for disposal in a geologic repository would decrease. For all alternatives under the proposed action, the volume of the new high-level radioactive waste forms would be less than the volume of untreated sodium-bonded spent nuclear fuel (the No Action Alternative). However, as stated in the Waste Management discussions, the projected amount of high-level radioactive waste would not require additional treatment and storage capacities beyond the current and planned INEEL capacities.

Table 4–67 lists the cumulative total waste generated at ANL-W and INEEL for years 2000 to 2035. The estimated quantity of radioactive/hazardous waste from baseline operations in this forecast through the year 2035 would be 205,550 cubic meters (7.25 million cubic feet). Waste generated by Alternative 6: Melt and dilute blanket and driver fuel at ANL-W (the alternative generating the most waste in all categories) would add a total of 980 cubic meters (34,610 cubic feet). During a 15-year time period, other reasonably foreseeable activities associated with the treatment of high-level radioactive waste and facility disposition at INEEL could add an additional 30,730 cubic meters (1.1 million cubic feet). Therefore, the potential cumulative total amount of waste generated from ANL-W and INEEL activities would be 237,260 cubic meters (8.4 million cubic feet).

Table 4–67 Estimated Cumulative Total Waste Generation for Years 2000 to 2035 From ANL-W and INEEL Concurrent Activities (Cubic Meters)

Waste Type	*ANL-W and INEEL Baseline Operations* [a]	*Idaho HLW and Facility Disposition EIS* [a]	*SBSNF EIS* [b]	*Total*
High-level radioactive	0	0	0 [d]	0 [c]
Low-level radioactive	135,600	15,320	925	151,845
Hazardous/mixed low-level radioactive	4,950	15,300	40	20,290
Transuranic	65,000[d]	110	15	65,125
Total	205,550	30,730	980	237,260

HLW = High-level radioactive waste.

[a] DOE 1999e: Figures 5.4–1 through 5.4–3 and input values for those figures through year 2035, Separations Alternative. Maximum quantities for any alternative.

[b] Alternative 6: Melt and dilute blanket and driver fuel at ANL-W; 12 years of operation. This alternative would generate the most waste in all categories.

[c] During treatment, the sodium-bonded spent nuclear fuel from existing inventories at ANL-W and INEEL would be converted into a high-level radioactive waste form for disposal in a geologic repository. For any alternative, the amount of material at ANL-W and INEEL scheduled for disposal in a geologic repository would not increase.

[d] In storage at the Radioactive Waste Management Complex.

The Central Facilities Area and Bonneville County landfill accepts nonhazardous and nonradioactive solid waste generated at INEEL. The onsite landfill complex was designed to accommodate combined ANL-W and INEEL solid waste disposal needs for a projected maximum operational life of 30 years.

The activities supporting the treatment and management of sodium-bonded spent nuclear fuel and other planned ANL-W and INEEL activities would not generate larger volumes of radioactive, hazardous, or solid waste beyond the current and projected capacities of ANL-W and INEEL waste storage and/or management facilities.

4.11.2 Savannah River Site

Nuclear facilities within a 80-kilometer (50-mile) radius of SRS include Georgia Power's Vogtle Electric Generating Plant across the river from SRS; Chem-Nuclear Inc., a commercial low-level radioactive waste burial site just east of SRS; and Starmet CMI, Inc. (formerly Carolina Metals), located southeast of SRS, which processes uranium-contaminated metals. Radiological impacts from the operation of the Vogtle Electric Generating Plant, a two-unit commercial nuclear power plant, are minimal, but DOE has factored them into the analysis. As stated in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000), the South Carolina Department of Health and Environmental Control Annual Report indicates that operation of the Chem-Nuclear Services facility and the Starmet CMI facility do not noticeably impact radiation levels in air or liquid pathways in the vicinity of SRS. Therefore, they are not included in this assessment.

The counties surrounding SRS have numerous existing (e.g., textile mills, paper product mills, and manufacturing facilities) and planned (e.g., Bridgestone Tire) industrial facilities with permitted air emissions and discharges to surface waters. Because of the distances between SRS and the private industrial facilities, there is little opportunity for interactions of plant emissions, and no major cumulative impact on air or water quality. Construction and operation of Bridgestone Tire and Hankook Polyester facilities could affect the regional socioeconomic cumulative impacts.

Additional offsite facilities with the potential to affect the nonradiological environment include South Carolina Electric and Gas Company's Urquhart Station. Urquhart Station is a three-unit, 250-megawatt, coal- and natural gas-fired steam electric plant on Beech Island, South Carolina, located north of SRS. Because

of the distance between SRS and the Urquhart Station and the regional wind direction frequencies, there is little opportunity for any interaction of plant emissions, and no significant cumulative impact on air quality.

DOE also evaluated the impacts from its own proposed future actions by examining impacts to resources and the human environment, as shown in NEPA documentation related to SRS (see Section 1.6). Additional NEPA documents related to SRS that are considered in the cumulative impacts section include the following:

Environmental Assessment for the Tritium Facility Modernization and Consolidation Project at the Savannah River Site (DOE 1998a). This environmental assessment addresses the impacts of consolidating the tritium activities currently performed in Building 232-H into the new Building 233-H and Building 234-H. Tritium extraction functions would be transferred to the Tritium Extraction Facility. The overall impact would be to reduce the tritium facility complex net tritium emissions by up to 50 percent. Another positive effect of this planned action would be to reduce the amount of low-level radioactive job-control waste. Effects on other resources would be negligible. Therefore, impacts from the environmental assessment have not been included in this cumulative impacts analysis.

Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy Stored at the Rocky Flats Environmental Technology Site (DOE 1998d). DOE proposes to process certain plutonium-bearing materials being stored at the Rocky Flats Environmental Technology Site. These materials are plutonium residues and scrub alloy remaining from nuclear weapons manufacturing operations formerly conducted by DOE at Rocky Flats. DOE has decided to remove the plutonium from certain residues that would be shipped from the Rocky Flats Environmental Technology Site to SRS for stabilization. The separated plutonium would be stored at SRS pending disposition decisions. Environmental impacts from using the F-Canyon to chemically separate the plutonium from the remaining materials at SRS are included in this section.

Draft and Final Environmental Impact Statements for the Construction and Operation of a Tritium Extraction Facility at the Savannah River Site (DOE 1998b, DOE 1999b). DOE proposes to construct and operate a Tritium Extraction Facility at SRS to provide the capability to extract tritium from commercial light water reactor targets and targets of similar design. The purpose of the proposed action and alternatives evaluated in the EIS is to provide tritium extraction capability to support reactor tritium production. Environmental impacts from the maximum processing option in this EIS are included in this section.

Defense Waste Processing Facility Supplemental Environmental Impact Statement (DOE 1994). The selected alternative in the Record of Decision was the completion and operation of the Defense Waste Processing Facility to immobilize high-level radioactive waste at SRS. The facility is currently processing sludge from SRS high-level radioactive waste tanks. However, SRS baseline data are not representative of full Defense Waste Processing Facility operational impacts, including the processing of salt and supernate from these tanks. Therefore, the Defense Waste Processing Facility data is listed separately.

In addition, the cumulative impacts analysis also includes the impacts from actions proposed in this SBSNF EIS. Risks to members of the public and site workers from radiological and nonradiological releases are based on operational impacts from the alternatives described in Sections 4.5 and 4.7, of this EIS.

Temporal limits were defined by examining the period of influence from both the proposed action and other Federal and nonfederal actions that have the potential for cumulative impacts. Actions to support the treatment of sodium-bonded blanket spent nuclear fuel at SRS are expected to begin in 2003 in preparation for ultimate offsite disposal, possibly in a geologic repository which probably will not be available until at least 2010. Final offsite shipments for spent nuclear fuel currently assigned to SRS for disposal would be completed by 2035.

The period of interest for the cumulative impacts analysis for this EIS includes the proposed construction and operation of facilities identified in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) and the Draft EIS for the SRS Tritium Extraction Facility (DOE 1998b, DOE 1999b), while actions for nuclear materials, highly enriched uranium, and surplus plutonium disposition would be ongoing.

4.11.2.1Air Resources

Table 4–68 compares the cumulative concentrations of nonradiological air pollutants from 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 SRS in all cases would be below the regulatory standards at the site boundary. Among the pollutants, the concentration of sulfur dioxide comes closest to the standard (approximately 96 percent of the standard for the 24-hour averaging time). The remaining pollutant emissions would range from 25 to 93 percent of the applicable standards.

Table 4–68 Estimated Maximum Cumulative Ground-Level Concentrations of Nonradiological Criteria Pollutants (Micrograms per Cubic Meter) at the SRS Boundary

Pollutant	*Averaging Time*	*Most Stringent Standard or Guideline* [a]	*SRS Baseline* [b]	*SBSNF EIS* [c]	*Other Foreseeable Planned SRS Activities* [d]	*Cumulative Concentrations* [e]
Carbon monoxide	8 hours	10,000 [f]	6,900	1.22	6.78	6,908
	1 hour	40,000 [f]	10,000	9.06	44.63	10,054
Nitrogen dioxide	Annual	100 [f]	26	3.11	4.63	34
PM_{10}	Annual	50 [f]	25	Less than 0.01	0.21	25
	24 hours (interim)	150 [f]	130	0.11	6.82	137
	24 hours (99^{th} percentile over 3 years)	150 [g]	(h)	Not available	Not available	Not available
$PM_{2.5}$	3-year annual	15 [g]	(h)	Not available	Not available	Not available
	24 hours (98^{th} percentile over 3 years)	65 [g]	(h)	Not available	Not available	Not available
Sulfur dioxide	Annual	80 [f]	34	Less than 0.01	0.06	34
	24 hours	365 [f]	350	0.12	0.96	351
	3 hours	1,300 [f]	1,200	0.91	5.28	1,206

PM_n = Particulate matter less than or equal to *n* microns in diameter.
[a] The more stringent Federal or state standard is presented if both exist for the averaging period.
[b] Data from Table 3–16 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data compiled from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5–1), including contributions from the Preferred Alternative less contributions from SBSNF EIS.
[e] Values in this column are rounded to the nearest number.
[f] Federal and state standard.
[g] Federal standard.
[h] No data available with which to assess particulate matter concentrations.

DOE also evaluated the cumulative impacts of airborne radiological releases in terms of dose to a maximally exposed offsite individual at the SRS boundary. DOE included the impacts of the Vogtle Plant (NRC 1996) in this cumulative total. The radiological emissions from the operation of the Chem-Nuclear low-level radioactive waste disposal facility just east of SRS are very low (DOE 2000) and are not included. **Table 4–69** lists the results of this analysis, using 1996 emissions (1992 for the Vogtle Plant) for the SRS baseline. The cumulative dose to the maximally exposed offsite individual would be 0.10 millirem per year, well below the regulatory standard of 10 millirem per year (40 CFR Part 61). Summing the doses to the

maximally exposed offsite individual for the proposed action and baseline SRS operations listed in Table 4–69 is an extremely conservative approach because, to get the calculated dose, the maximally exposed offsite individual would have to occupy different physical locations at the same time, which is impossible.

Table 4–69 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Offsite Population in the 80-Kilometer (50-Mile) Radius From Airborne Releases at SRS

Activity	*Maximally Exposed Offsite Individual*		*Population* [a]	
	Dose (millirem)	*Latent Cancer Fatality Risk*	*Collective Dose (person-rem)*	*Excess Latent Cancer Fatalities*
SRS baseline [b]	0.050	2.5×10^{-8}	5.5	0.0028
SBSNF EIS [c]	0.00039	2.0×10^{-10}	0.019	9.5×10^{-6}
Management of spent nuclear fuel [d]	0.015	7.5×10^{-9}	0.56	0.00028
Disposition of surplus highly enriched uranium [d]	0.0025	1.3×10^{-9}	0.16	0.00008
Tritium Extraction Facility [d]	0.02	1.0×10^{-8}	0.77	0.00039
Disposition of surplus plutonium [d]	0.0074	3.7×10^{-9}	1.8	0.0009
Management of plutonium residues/scrub alloy [d]	0.00057	2.9×10^{-10}	0.0062	3.1×10^{-6}
Defense Waste Processing Facility [d]	0.001	5.0×10^{-10}	0.071	0.000036
DOE Complex miscellaneous components [d]	0.0044	2.2×10^{-9}	0.007	3.3×10^{-6}
Vogtle Plant [d]	0.00054	2.7×10^{-10}	0.042	0.000021
Total	0.10	5.1×10^{-8}	8.94	0.0045

[a] A collective dose to the 80-kilometer (50-mile) population.
[b] Data from Table 3–21 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5–2 maximum impact alternative).

Adding the population doses from current and projected activities at SRS, the Vogtle Plant, the SRS Spent Nuclear Fuel Management Final EIS, and this EIS could yield a total annual cumulative dose of 8.94 person-rem from airborne sources. The total annual cumulative dose translates into 0.0045 latent cancer fatalities for each year of exposure for the population living within an 80-kilometer (50-mile) radius of SRS.

4.11.2.2Water Resources

At present, a number of SRS facilities discharge treated wastewater to Upper Three Runs and its tributaries and Fourmile Branch via NPDES–permitted outfalls. These include the F- and H-Area Effluent Treatment Facility and the M-Area Liquid Effluent Treatment Facility. As stated in Sections 4.5.2 and 4.7.2, operations associated with the treatment and management of sodium-bonded spent nuclear fuel are not expected to result in any discharges to groundwater. The only technology that would result in discharges of radiological and nonradiological effluent to surface water would be PUREX processing. The major sources of liquid effluent from facilities associated with PUREX processing would be process cooling water and steam condensate systems that could contain small quantities of radionuclides and chemicals. This process wastewater would be treated at the F-Area Effluent Treatment Facility and then discharged to the Upper Three Runs. Studies of water quality and biota downstream of the Effluent Treatment Facility outfall suggest that discharges have not degraded the water quality of Upper Three Runs (DOE 2000).

Other potential sources of contaminants into Upper Three Runs during the periods in which sodium-bonded spent nuclear fuel would be treated in F-Area using PUREX, or in L-Area using melt and dilute treatment,

include activities described in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000), the tritium extraction facility, environmental restoration, and decontamination and deactivation activities, as well as modifications to existing SRS facilities. Discharges from activities associated with the SRS Spent Nuclear Fuel Management Final EIS and tritium extraction facility would not add significant amounts of nonradiological contaminants to Upper Three Runs. The amount of discharge associated with environmental restoration and decontamination and deactivation activities would vary based on the level of activity. All the potential activities that could result in wastewater discharges would be required to comply with the NPDES permit limits that ensure protection of water quality.

Table 4–70 summarizes the estimated cumulative radiological doses from waterborne sources to human receptors downstream from SRS. Liquid effluent released to SRS streams that are tributaries of the Savannah River could contain small quantities of radionuclides. The exposure pathways considered in this analysis included drinking water, fish ingestion, shoreline exposure, swimming, and boating. The estimated cumulative dose to the maximally exposed offsite individual from liquid releases would be 0.24 millirem per year, well below the regulatory standard of 4 millirem per year (40 CFR Part 141). Adding the population doses associated with current and projected SRS activities would yield a cumulative annual dose of 2.6 person-rem from liquid sources. This translates into 0.0013 latent cancer fatalities for each year of exposure of the population living within an 80-kilometer (50-mile) radius of SRS.

Table 4–70 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Offsite Population in the 80-Kilometer (50-Mile) Radius From Liquid Releases at SRS

	Maximally Exposed Offsite Individual		*Population* [a]	
Activity	*Dose (millirem)*	*Latent Cancer Fatality Risk*	*Collective Dose (person-rem)*	*Excess Latent Cancer Fatalities*
SRS baseline [b]	0.13	6.5×10^{-8}	2.4	0.0012
SBSNF EIS [c]	0.00012	6.0×10^{-11}	0.00068	3.4×10^{-7}
Management of spent nuclear fuel [d]	0.057	2.9×10^{-8}	0.19	0.000095
Disposition of surplus highly enriched uranium [d]	(e)	(e)	(e)	(e)
Tritium Extraction Facility [d]	(e)	(e)	(e)	(e)
Defense Waste Processing Facility [d]	(e)	(e)	(e)	(e)
Disposition of surplus plutonium [d]	(e)	(e)	(e)	(e)
Management plutonium residues/scrub alloy [d]	(e)	(e)	(e)	(e)
DOE Complex miscellaneous components [d]	0.000042	2.1×10^{-11}	0.00024	1.2×10^{-7}
Plant Vogtle [d]	0.054	2.7×10^{-8}	0.0025	1.3×10^{-6}
Total	0.24	1.2×10^{-7}	2.6	0.0013

[a] A collective dose to the 80-kilometer (50-mile) and downstream population.
[b] Data from Table 3–21 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5-3 maximum impact alternative).
[e] Less than minimum reportable levels.

4.11.2.3Socioeconomic Impacts

No additional workers would be required for the operation of PUREX and melt and dilute facilities at SRS as a result of the proposed action. Therefore, there would be no contribution to the cumulative impacts.

4.11.2.4Public and Worker Health

Table 4–71 summarizes the cumulative radiological health effects of routine SRS operations, proposed DOE actions, and nonfederal nuclear facility operations (Vogtle Electric Generating Plant). Impacts resulting from proposed DOE actions are described in the EISs listed previously in this chapter. In addition to estimated radiological doses to the hypothetical maximally exposed offsite individual, the offsite population, and the workers, Table 4–71 lists the potential number of latent cancer fatalities for the public and workers due to radiation exposure. The radiation dose to the maximally exposed offsite individual from air and liquid pathways would be 0.34 millirem per year, which is well below the applicable DOE regulatory limits (10 millirem per year from the air pathway; 4 millirem per year from the liquid pathway; and 100 millirem per year for all pathways). The total annual population dose for current and projected activities of 11.5 person-rem translates into 0.0058 latent cancer fatalities for each year of exposure for the population living within an 80-kilometer (50-mile) radius of SRS.

The annual radiation dose to a worker population would be 859 person-rem. In addition, doses to individual workers would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1, which is well below the regulatory limit of 5,000 millirem per year (10 CFR 835). Furthermore, standards and practices to ensure worker doses are as low as reasonably achievable would be exercised to maintain individual worker doses below the DOE Administrative Control Level of 2,000 millirem per year.

4.11.2.5Environmental Justice

As discussed in Chapter 4 and Appendix H, implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel at SRS would have no significant impact on public health or the environment. Therefore, the implementation of either of two alternatives at SRS would result in no disproportionately high and adverse impacts on minority or low-income populations residing within potentially affected areas.

4.11.2.6Waste Generation

As stated in Sections 4.5.6 and 4.7.6, high-level and low-level radioactive waste, transuranic waste, and hazardous/mixed waste would be generated from the treatment of declad and cleaned sodium-bonded blanket spent nuclear fuel at SRS. The largest volumes of low-level radioactive and transuranic waste would be generated with PUREX processing. However, as stated in Sections 4.5.6 and 4.7.6, the projected waste generation rates would not require additional treatment and storage capacities beyond the current and planned SRS capacities. It should be noted that the treatment of blanket spent nuclear fuel at SRS would result in the generation of new high-level radioactive waste that would be added to the SRS current inventory. This is because the blanket spent nuclear fuel would be transported from ANL-W to SRS for treatment.

Table 4–72 lists the cumulative volumes of liquid high-level and solid low-level radioactive, transuranic, and hazardous/mixed waste that SRS would generate. The table includes data from the SRS 30-year expected waste forecast. The 30-year expected waste forecast is based on operations, environmental restoration, and decontamination and deactivation waste forecasts from existing generators and the following assumptions: (1) secondary waste from the Defense Waste Processing Facility, In-Tank Precipitation, and Extended Sludge Processing operations are addressed in the Defense Waste Processing Facility EIS; (2) high-level radioactive waste volumes are based on the selected option for the F-Canyon Plutonium Solutions EIS; (3) some investigation-derived waste is handled as hazardous waste per RCRA regulations; (4) purge water from well samplings is handled as hazardous waste; and (5) the continued receipt of small amounts of low-level radioactive waste from other DOE facilities and nuclear naval operations (DOE 2000).

Table 4–71 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Offsite Population and Facility Workers at SRS

Activity	*Maximally Exposed Offsite Individual*				*Population* [a]				*Workers*	
	Dose From Airborne Releases (millirem)	*Dose From Liquid Releases (millirem)*	*Total Dose (millirem)*	*Latent Cancer Fatality Risk*	*Dose From Airborne Releases (person-rem)*	*Dose From Liquid Releases (person-rem)*	*Collective Dose (person-rem)*	*Excess Latent Cancer Fatalities*	*Collective Dose (person-rem)*	*Excess Latent Cancer Fatalities*
SRS baseline [b]	0.050	0.13	0.18	9.5×10^{-8}	5.5	2.4	7.9	0.0025	165	0.066
SBSNF EIS [c]	0.00039	0.00012	0.00051	2.6×10^{-10}	0.019	0.00068	0.020	1.0×10^{-8}	38	0.015
Management of spent nuclear fuel [d]	0.015	0.057	0.072	3.6×10^{-8}	0.56	0.19	0.75	0.00038	55	0.022
Disposition of surplus highly enriched uranium [d]	0.0025	(e)	0.0025	1.3×10^{-8}	0.16	(e)	0.16	0.00008	11	0.00044
Tritium Extraction Facility [d]	0.02	(e)	0.02	1.0×10^{-8}	0.77	(e)	0.77	0.00039	4	0.0016
Defense Waste Processing Facility [d]	0.001	(e)	0.001	5.0×10^{-10}	0.071	(e)	0.071	0.000036	120	0.048
Disposition of surplus plutonium [d]	0.0074	(e)	0.0074	3.7×10^{-9}	1.8	(e)	1.8	0.0009	456	0.18
Management plutonium residues/ scrub alloy [d]	0.00057	(e)	0.00057	2.9×10^{-10}	0.0062	(e)	0.0062	3.1×10^{-6}	7.6	0.003
DOE Complex miscellaneous components [d]	0.0044	0.000042	0.0044	2.2×10^{-9}	0.007	0.00024	0.0072	3.6×10^{-6}	2	0.001
Vogtle Plant [d]	0.00054	0.054	0.055	2.7×10^{-8}	0.042	0.0025	0.045	0.000022	Not available	
Total	0.10	0.24	0.34	1.9×10^{-7}	8.95	2.60	11.5	0.0058	859	0.34

[a] A collective dose to the 80-kilometer (50-mile) population for atmospheric releases and to the downstream users of the Savannah River for liquid releases.
[b] Data from Tables 3–21 and 3–22 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5-4).
[e] Less than minimum reportable levels.

Table 4–72 Estimated Cumulative Total Waste Generation From SRS Concurrent Activities (Cubic Meters)

Waste Type	*SRS Baseline Operations* [a]	*SBSNF EIS* [b]	*Spent Nuclear Fuel Management* [a]	*ER/D&D* [a]	*Other Waste Volume* [a]	*Total* [c]
Liquid high-level radioactive	14,129	510	11,000	0	69,040	94,680
Low-level radioactive	118,669	900	140,000	61,630	109,200	430,400
Hazardous/mixed low-level radioactive	3,856	7	270	6,173	4,430	14,740
Transuranic	6,012	90	3,700	0	8,730	18,530
Total [c]	142,670	1,510	154,970	67,800	191,400	558,350

ER/D&D = environmental restoration/decontamination and decommissioning; based on a total 30-year expected waste forecast, including previously generated waste.

[a] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) maximum impact alternative, Table 5-5, based on a total 30-year expected waste forecast, including previously generated waste, and adjusted for the SBSNF EIS.

[b] Alternative 3: PUREX processing of declad and cleaned blanket spent nuclear fuel at SRS F-Canyon.

[c] The values are rounded to the nearest ten; the total sum may be different from the sum of individuals.

As indicated in Table 4–72, the estimated quantity of radioactive/hazardous waste from SRS operations in this forecast during the next 30 years would be approximately 142,670 cubic meters (5.04 million cubic feet). Waste generated by Alternative 3: PUREX processing of blanket fuel at SRS F-Canyon, would add a total of approximately 1,510 cubic meters (53,330 cubic feet). Waste generated from the conventional (PUREX) processing option described in the SRS Spent Nuclear Fuel Management Final EIS would add a total of 154,970 cubic meters (5.48 million cubic feet). In addition, radioactive/hazardous waste associated with environmental restoration and decontamination and decommissioning activities would have a 30-year expected forecast of approximately 67,800 cubic meters (2.39 million cubic feet) (DOE 2000). During this same time period, other reasonably foreseeable activities that were not included in the 30-year forecast would add, approximately, an additional 191,400 cubic meters (6.76 million cubic feet). Therefore, the potential cumulative amount of waste generated from SRS activities during the period of interest would be approximately 558,350 cubic meters (19.72 million cubic feet). It is important to note that the quantities of waste generated are not equivalent to the amounts that will require disposal. At SRS, high-level radioactive material is evaporated and concentrated to a smaller volume for final disposal. Combustible low-level radioactive waste is volume-reduced on site in the Consolidated Incineration Facility.

The Three Rivers Solid Waste Authority Regional Waste Management Center at SRS accepts nonhazardous and nonradioactive solid waste from SRS and eight surrounding South Carolina counties. This municipal solid waste landfill provides state-of-the-art Subtitle D (nonhazardous) facilities for land-filling solid waste while reducing the environmental consequences associated with construction and operation of multiple county-level facilities. It was designed to accommodate combined SRS and county solid waste disposal needs for at least 20 years, with a projected maximum operational life of 45 to 60 years (DOE 2000). The landfill is designed to handle an average of 1,000 tons per day and a maximum of 2,000 tons per day of municipal solid waste. The SRS and eight cooperating counties had a combined generation rate of 900 tons per day in 1995. The Three Rivers Solid Waste Authority Regional Waste Management Center opened in mid-1998.

Activities supporting the treatment and management of sodium-bonded spent nuclear fuel and other planned SRS activities would not generate larger volumes of radioactive, hazardous, or solid waste beyond the current and projected capacities of SRS waste storage and/or management facilities.

4.11.3 Transportation

The Programmatic Spent Nuclear Fuel EIS (DOE 1995a) analyzed the cumulative impacts of all transportation of radioactive materials, including impacts from reasonably foreseeable future actions that include transportation of radioactive material for a specific purpose and general radioactive materials transportation that is not related to a particular action. The total worker and general population collective doses are expected to be less than 1 person-rem. The impacts of this program are quite small compared with overall transportation impacts. Total collective worker doses from all types of shipments (historical, the alternatives, reasonably foreseeable actions, and general transportation) were estimated to be 320,000 person-rem (130 latent cancer fatalities) for the period 1943 through 2035 (93 years).

4.12 PROGRAMMATIC CONSIDERATIONS

Programmatic considerations presented in this section provide information on the regulatory environment applicable to spent nuclear fuel and high-level radioactive waste. Also presented are schedule considerations for the disposal of DOE-owned spent nuclear fuel or high-level radioactive waste in a geologic repository.

4.12.1 Regulatory Environment Considerations

Prior to the acceptance of spent nuclear fuel or high-level radioactive waste at a geologic repository, certain NRC and EPA regulatory requirements and DOE Office of Civilian Radioactive Waste Management requirements must be met. Regulatory requirements specific to DOE's sodium-bonded spent nuclear fuel, are identified in the Civilian Radioactive Waste Management Office's current April 1999, Draft Waste Acceptance System Requirements Document (DOE 1999c).

One of the key NRC requirements for acceptance of spent nuclear fuel or high-level radioactive waste is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive (in the repository environment) in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective [10 CFR 60.135(b)(1)]. The No Action Alternative may not satisfy this requirement, because the metallic sodium is highly reactive. The metallic uranium is also reactive and potentially pyrophoric, and in some cases the fuel contains highly enriched uranium, which would require criticality control measures. It also is uncertain whether the treatment technology, identified for the blanket spent nuclear fuel under Alternative 2 (cleaning the fuel to remove sodium and packaging in a high-integrity can), would be adequate to meet this NRC requirement. Under all other action alternatives, this requirement could be met.

The Waste Acceptance System Requirements Document identifies the Civilian Radioactive Waste Management Office's acceptance criteria for spent nuclear fuel and high-level radioactive waste. Under these criteria, the Civilian Radioactive Waste Management Office states that only spent nuclear fuel and high-level radioactive waste that is not subject to regulation under RCRA, Subtitle C, will be accepted for disposal. Untreated sodium-bonded spent nuclear fuel may be regulated under RCRA, Subtitle C, because it exhibits the characteristic of reactivity when exposed to water (40 CFR 261.23 (a)(2), (3)) and is ignitable (40 CFR 261.21 (a)(2)).

Under RCRA, 40 CFR 268.9 (c), "... no prohibited waste which exhibits a characteristic under 40 CFR part 261, subpart C, may be land disposed of unless the waste complies with the treatment standards under 40 CFR 268, subpart D." Deactivation is the waste treatment technology for waste that exhibits the characteristic of reactivity and ignitability (40 CFR 268.40). RCRA land disposal requirements (i.e., 40 CFR 268.40) also require generators of waste that exhibits the characteristics of reactivity to water or ignitability to identify all underlying hazardous constituents reasonably expected to be present in the waste at the point of generation, and to treat these constituents to the universal treatment standards. If the characteristic waste

is treated by the applicable treatment technology and the waste no longer exhibits the characteristic, then the waste no longer needs to be regulated under RCRA, Subtitle C, and can be managed as a nonhazardous waste (62 *FR* 62083).

The direct disposal option of the No Action Alternative may not satisfy this requirement, because the sodium-bonded spent nuclear fuel could be considered to be reactive and ignitable, and therefore, it may not be accepted for disposal at a geologic repository. All of the alternatives under the proposed action would be able to deactivate the sodium-bonded fuel and remove the characteristics of reactivity and ignitability. The metallic uranium is considered to be reactive, and in some cases pyrophoric; however, it would not be a RCRA hazardous waste because it is defined under the Atomic Energy Act of 1954, as amended (42 U.S.C. 2001 *et seq*.), as a source, special nuclear, or byproduct material and, therefore, is excluded from RCRA under 40 CFR 261.4 (a)(4).

The Waste Acceptance System Requirements Document also identifies specific acceptance criteria for DOE's spent nuclear fuel and high-level radioactive waste. For high-level radioactive waste, the Civilian Radioactive Waste Management Office specifies borosilicate glass as a standard vitrified high-level radioactive waste form. For DOE's spent nuclear fuel, specific acceptance criteria have been developed for canistered DOE spent nuclear fuel, including naval spent nuclear fuel that is intended for disposal in the canister. Performance criteria for the ceramic high-level radioactive waste and the metallic high-level radioactive waste forms are being developed. However, no specific acceptance criteria have been developed for spent nuclear fuel that has been melted into a liquid form and then solidified. The No Action Alternative may be able to meet this requirement for the disposal canisters; however, as previously discussed, it may not meet the other waste acceptance requirements (e.g., NRC and RCRA regulations).

For Alternative 3 (blanket fuel) where the treated waste form would be a vitrified borosilicate glass, the specific acceptance criteria have been developed. However, final approval of this waste form would be contingent upon the requirements in the disposal facility's license.

For Alternative 1 (blanket and driver fuel), Alternative 2 (driver fuel), Alternative 3 (driver fuel), Alternative 4 (driver fuel), and Alternative 5 (driver fuel), performance criteria for the ceramic high-level radioactive waste and the metallic high-level radioactive waste form have been developed, but need approval. Again, final approval of this waste form would be contingent upon the requirements in the disposal facility's license.

For Alternative 2 (blanket fuel), the specific acceptance criteria for canistered spent nuclear fuel would apply and may be achieved. However, the long-term durability of the proposed overpack container has not been demonstrated or documented. Without such demonstration of extended containment, the ability of the high-integrity can concept to meet the safety standards proposed by the National Research Council is unknown (National Research Council 1998).

For Alternative 4 (blanket fuel), Alternative 5 (blanket fuel), and Alternative 6 (blanket and driver fuel), the specific acceptance criteria for conditioned spent nuclear fuel would need to be developed and approved.

4.12.2 Schedule Considerations

The schedule perspective for each of the alternatives is affected by two time frames: the proposed schedule for the construction, operation, and closure of a geologic repository, and 2035, the year by which DOE committed to remove all spent nuclear fuel from Idaho under the 1995 agreement with the State of Idaho.

The proposed schedule for the repository is discussed in the *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998f). A site recommendation decision for the geologic repository is scheduled for 2001. If the site were to be subsequently authorized, a license application could be submitted in 2002. The NRC construction authorization decision could occur in 2005 at the earliest. Repository construction would begin upon receipt of this authorization. DOE must update its licensing application and submit it to the NRC before the Commission will issue a license to receive and process nuclear waste. This update is scheduled for 2008. Assuming repository construction sufficient to begin waste emplacement will take five years, the first waste emplacement at Yucca Mountain could occur in 2010. DOE would design the repository to close as early as approximately 10 years after emplacement of the last waste package, or to be kept open for at least 100 years after initiation of waste emplacement, with a reasonable expectation that the repository actually could be kept open with appropriate maintenance for 300 years after initiation of waste emplacement. The Viability Assessment (DOE 1998f) assumes a reference case in which closure of a monitored geologic repository is initiated in 2110, 100 years after initiation of waste emplacement operations.

Under the No Action Alternative, the untreated sodium-bonded spent nuclear fuel could remain in storage at the current locations until 2035. After that, it would need to be transported outside the State of Idaho and stored or treated at another DOE site. If the waste acceptance criteria are finalized by 2010 and indicate that direct disposal of the sodium-bonded spent nuclear fuel is possible, the fuel could be packaged for direct disposal well before 2035.

The treatment of the driver spent nuclear fuel using the electrometallurgical technology under Alternatives 1 through 5 could start as early as 2000 and could be completed by 2006 to 2007. If the decision to select a technology is delayed until after 2010, when waste acceptance criteria may be finalized, it would require two to three years lead time for the reactivation or installation of new equipment for the electrometallurgical treatment technology and six to seven years for the processing, for a total of approximately 10 years. The high-level radioactive waste would be ready for disposal by 2020.

The treatment of driver spent nuclear fuel only using the melt and dilute process at ANL-W could start as early as 2005 and could be completed by 2007. If installation of the necessary equipment is delayed until after 2010, the conditioned spent nuclear fuel would be ready for disposal in 2017.

The treatment of the blanket spent nuclear fuel using the electrometallurgical technology under Alternative 1 could start as early as 2000 and could be completed by 2012 or 2013. Delaying a decision until after 2010 would add 10 to 15 years, depending on the time required to reactivate or install new equipment. The process still could be completed by 2030.

The preparation of the blanket spent nuclear fuel and its placement in high-integrity cans under Alternative 2 could start in 2003. Cleaning and sodium removal activities and packaging would take approximately six years and could be completed by 2009. Delaying a decision until after 2010 would delay the completion of this effort to approximately 2020.

The treatment of blanket spent nuclear fuel using the PUREX process at SRS would not start until 2005 because the F-Canyon is committed to other missions. Once started, however, all blanket spent nuclear fuel could be processed in less than one year. The decladding and sodium removal activities at ANL-W to prepare the blanket spent nuclear fuel for transportation and processing also would not start until 2003, allowing for installation of new equipment. It is estimated that preparation activities at ANL-W for all blanket spent nuclear fuel would last approximately six years. Therefore, the overall process could be completed by approximately 2010. At this time it is not clear whether the decision to process blanket spent nuclear fuel at the F-Canyon could be delayed until after 2010 because DOE has scheduled operations for the F-Canyon until 2005. If there were a programmatic decision to close the F-Canyon after currently scheduled operations are completed, the F-Canyon would not be available.

The preparation of blanket spent nuclear fuel for the melt and dilute process at ANL-W under Alternative 4 could start in 2003, allowing time for the installation of new equipment. The melt and dilute activities could start in 2005 and could be completed in seven years (by 2012). The process would require sodium removal activities at ANL-W, which could be done in parallel. The blanket spent nuclear fuel preparation activities would start in 2003 and would require approximately six years for completion. The overall process could be completed by 2012. If a decision were delayed until after 2010, treatment would not be completed until about 2020.

The treatment of blanket spent nuclear fuel using the melt and dilute process at SRS under Alternative 5 could start after 2020 if capacity becomes available. It is estimated that the facility would be operational by 2005, but it is committed to other missions until 2035, as stated in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). If additional capacity becomes available, treatment could start as soon as 2020. Processing of the blanket spent nuclear fuel at SRS would take approximately three years. The decladding and sodium removal activities at ANL-W that are needed to prepare the fuel could start in 2003 and could be completed by 2009, well before processing begins. Delaying a decision until 2010 would push the completion of the decladding activities to 2019, which would be well before processing could begin at SRS.

The treatment of blanket and driver spent nuclear fuel using the melt and dilute process at ANL-W under Alternative 6 could start as early as 2003 and could be completed by 2015. Delaying a decision until 2010 would push completion to approximately 2025.

Table 4–73 summarizes the dates for completing the process for each alternative, given that a decision to proceed is made in the year 2000 or the year 2010.

Table 4–73 Treatment Completion Year

Alternatives [a]	*Decision in 2000*	*Decision in 2010*
No Action (direct disposal)	Before 2035	Before 2035
Alternative 1		
Driver (only)	2006	2020
Driver and blanket	2012	2030
Alternative 2		
Driver	2006	2020
Blanket	2009	2020
Alternative 3		
Driver	2006	2020
Blanket	2010	F-Canyon may not be available
Alternative 4		
Driver	2006	2020
Blanket	2012	2020
Alternative 5		
Driver	2006	2020
Blanket	2025	2025
Alternative 6		
Driver (only)	2007	2017
Driver and blanket	2015	2025

[a] See Section 2.5 for an explanation of alternatives.

4.13 MITIGATION MEASURES

Following completion of an EIS and its associated Record of Decision, DOE is required to prepare a Mitigation Action Plan to address any mitigation commitments expressed in the Record of Decision (10 CFR

1021.331). The purpose of the Mitigation Action Plan is to explain how measures designed to mitigate adverse environmental impacts will be planned and implemented. The Mitigation Action Plan is prepared prior to DOE taking any action directed by the Record of Decision that is the subject of a mitigation commitment.

Based on analyses of the environmental consequences of the proposed action presented earlier in this chapter, no mitigation measures would be necessary since all potential environmental impacts would be small and well within applicable requirements. Each DOE site would follow installation and operational practices that would minimize any potential impacts to air and surface water quality, noise, operational and public health and safety, and accident prevention and mitigation. These practices are dictated by Federal and state licensing and permitting requirements, as described in Chapter 5.

4.14 RESOURCE COMMITMENTS

This section describes the unavoidable adverse environmental impacts that could result from the proposed action; the relationship between short-term uses of the environment and the maintenance and enhancement of long-term productivity; and irreversible and irretrievable commitments of resources. Unavoidable adverse environmental impacts are impacts that would occur after implementation of all feasible mitigation measures. The relationship between short-term uses of the environment and the maintenance and enhancement of long-term productivity addresses issues associated with the condition and maintenance of existing environmental resources used to support the proposed action and the utility of these resources after their use. Resources that would be irreversibly and irretrievably committed are those that cannot be recovered or recycled and those that are consumed or reduced to unrecoverable forms.

4.14.1 Unavoidable Adverse Environmental Impacts

Implementing any of the alternatives considered in this EIS for the treatment and management of sodium-bonded spent nuclear fuel would result in unavoidable adverse impacts to the human environment. In general, these impacts are expected to be minimal and would come from incremental impacts attributed to the operation of treatment and management facilities at ANL-W and SRS.

Operation of treatment and management facilities at ANL-W and SRS would result in unavoidable increases of radiation exposures to workers and the general public. Workers would be exposed to direct radiation and other chemicals associated with the handling and treatment of the sodium-bonded spent nuclear fuel. The incremental annual dose contribution from the treatment and management of sodium-bonded spent nuclear fuel to the maximally exposed offsite individual, general population, and workers are discussed in Sections 4.3.4, 4.4.4, 4.5.4, 4.6.4, 4.7.4, and 4.8.4.

Also unavoidable would be the generation of additional low-level transuranic and mixed radioactive waste compared to baseline generation rates, which would either be treated and stored on site at ANL-W or SRS, or transported and managed off site at appropriate waste disposal facilities. Any other waste generated during treatment and management activities would be collected at the site, treated and/or stored, and eventually removed for suitable recycling or disposal off site in accordance with applicable EPA regulations.

Operation of treatment and management facilities at ANL-W and SRS would have minimal unavoidable adverse environmental impacts to air and water quality. Air quality would be affected by increases in various chemical or radiological constituents in the routine emissions typical of facility operations at these sites. Impacts to water resources and quality also would be affected by the release of various chemical or radiological constituents in the routine effluent only from PUREX processing at SRS. Impacts to the environment associated with the normal operation of facilities at ANL-W and SRS would occur regardless

of the treatment and management of spent nuclear fuel. These routine impacts also have been addressed in various other NEPA documentation at these sites.

The alternative treatment processes would generate varying amounts of waste material that could affect storage requirements. This would be an unavoidable impact on the amount of available and anticipated storage space and the requirements of disposal facilities.

4.14.2 Relationship Between Local Short-Term Uses of the Environment and the Maintenance and Enhancement of Long-Term Productivity

Implementation of the alternatives, including the No Action Alternative, would cause short-term commitments of resources (e.g., air emissions and water discharges) and would permanently commit certain resources (e.g., dilution materials and energy). For each alternative, the short-term use of these resources would result in potential long-term benefits to the environment and the enhancement of long-term productivity by decreasing overall health risks to workers, the public, and the surrounding environment by reducing their exposure to hazardous and radioactive substances. The short-term effect on workers, the public, and the environment from the treatment of sodium-bonded spent nuclear fuel would be offset by the long-term benefits of safe, stable, secure storage of these materials.

Under the No Action Alternative, environmental resources already have been committed to the storage of spent nuclear fuel. This commitment would serve to maintain existing environmental conditions with little or no impacts to the long-term productivity of the environment. The continued storage of sodium-bonded spent nuclear fuel at ANL-W and INEEL until 2035 and the potential for its direct disposal in a geologic repository would result in less exposure to hazardous and radioactive materials for workers, the public, and the environment than would be experienced under the proposed action. Only the direct disposal of the sodium-bonded fuel in a repository would have the potential to enhance the long-term viability of the environment in Idaho.

Under the proposed action, the short-term use of environmental resources at ANL-W and SRS would be greater than for the No Action Alternative. The short-term commitment of resources would include the space required for onsite processing, the commitment of processing facilities, transportation, and other disposal resources and materials for the treatment and management of sodium-bonded spent nuclear fuel. Workers, the public, and the environment would be exposed to larger amounts of hazardous and radioactive materials over the short-term from the handling and treatment of the spent nuclear fuel, including process emissions and the handling of waste. Again, these commitments would be offset by an even greater potential for enhanced long-term viability of the environment than under the No Action Alternative.

Over the life of the proposed action, groundwater would be used at SRS to meet sanitary and process needs. After use and treatment, this water would be discharged into surface water streams. Depending on the site chosen (F- or L-Area) and the technology implemented over the short-term, the resulting increases in pollutant loadings would take advantage of the natural assimilative capacity of the receiving stream(s). However, these incremental pollutant loadings should not adversely affect either short- or long-term viability of the aquatic ecosystem. These impacts would be assessed during the regulatory permitting process once an alternative has been selected.

Regardless of location, air emissions associated with implementation of any of the technologies would add small amounts of radiological and nonradiological constituents to the air of the regions around ANL-W and SRS. During the project's life, these emissions would result in additional loading and exposure, but would not impact compliance with air quality or radiation exposure standards at either site. There would be no significant residual environmental effects to long-term environmental viability.

The management and disposal of sanitary solid waste and nonrecyclable radiological waste over the project's life would require energy and space at ANL-W and SRS treatment, storage, or disposal facilities. The land required to meet the solid waste needs would require a long-term commitment of terrestrial resources. Upon the facilities' closures, DOE could decontaminate and decommission the facilities and/or equipment and restore them to brown field sites which could be available for future commercial or industrial development.

Regardless of location, continued employment, expenditures, and tax revenues generated during the implementation of any of the alternatives would directly benefit the local, regional, and state economies over the short-term. Long-term economic productivity could be facilitated by local governments investing project-generated tax revenues into infrastructure and other required services.

The use of short-term resources to operate spent nuclear fuel treatment and management facilities at either ANL-W or SRS would not affect the long-term productivity of these sites.

4.14.3 Irreversible and Irretrievable Commitments of Resources

Irreversible and irretrievable commitments of resources for each alternative potentially would include mineral resources during the life of the project and energy used in treating the waste. The commitment of capital, energy, labor, and material during the implementation of the alternatives generally would be irreversible.

Energy expended would be in the form of fuel for equipment and vehicles, electricity for facility operations, and human labor. The energy consumption of treatment and management facilities would be a small fraction of the total energy used at each DOE site. None of the technologies evaluated in the EIS would require significantly higher or lower energy consumption. Assuming that these facilities are totally dedicated to the treatment and management of sodium-bonded spent nuclear fuel, it is estimated that total electrical energy consumption would range from 101,500 megawatt hours for Alternative 2, high-integrity cans, to 130,000 megawatt hours for Alternative 5, melt and dilute at SRS. Operation of any proposed facility would generate nonrecyclable waste streams, such as radiological and nonradiological solid waste and some process wastewaters. However, certain materials and equipment used during operation of the proposed facility could be recycled when the facility is decontaminated and decommissioned.

The implementation of the alternatives considered in this EIS, including the No Action Alternative, would require water, electricity, steam, and diesel fuel. Water at SRS and ANL-W would be obtained from onsite groundwater sources and steam from existing onsite sources. Electricity and diesel fuel would be purchased from commercial sources. These commodities are readily available and the amounts required would not have an appreciable impact on available supplies or capacities. From a materials and energy resource commitment perspective, electrometallurgical treatment and PUREX process technologies would recover low-enriched uranium, which is usable as commercial reactor fuel.

The disposal of hazardous and/or radioactive waste also would cause irreversible and irretrievable commitments of land, mineral, and energy resources. Hazardous waste and low-level radioactive waste disposal would irreversibly and irretrievably commit land for its disposal. For each of the alternatives analyzed in this document, the No Action Alternative would have the least commitment of land, mineral, and energy resources.

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5. ENVIRONMENTAL LAWS, REGULATIONS, AND CONSULTATIONS

Chapter 5 presents the laws, environmental regulations, and consultations that apply to the proposed action and alternatives. Federal, state, and U.S. Department of Energy environmental, safety, and health laws, regulations, and directives are summarized in Section 5.1. Radioactive material packaging and transportation regulations are discussed in Section 5.2. Emergency management and response laws, regulations, and Executive Orders are discussed in Section 5.3. Consultations with Federal, state, and local agencies and Federally recognized Native American groups are discussed in Section 5.4.

5.1 ENVIRONMENTAL, SAFETY, AND HEALTH LAWS, REGULATIONS, EXECUTIVE ORDERS, AND DOE ORDERS

There are a number of Federal environmental laws dealing with environmental protection, compliance, or consultation that affect compliance at every U.S. Department of Energy (DOE) location. In addition, certain environmental requirements have been delegated to state authorities for enforcement and implementation. It is DOE policy to conduct its operations in a manner that ensures protection of public health, safety, and the environment through compliance with all applicable Federal and state laws, regulations, Orders, and other requirements. This chapter describes the environmental, safety, and health laws, regulations, and Executive and DOE Orders that are important to DOE's implementation of the proposed action. The applicability of these laws, regulations, and Orders and how they affect the proposed action are discussed in Chapters 1, 2, 3, and 4, and the appendices, where appropriate. Appendix B discusses regulations that pertain to the methodologies used in the environmental impact statement (EIS) analyses. Appendices E and F discuss applicable health and safety regulations. Appendix F discusses relevant transportation regulations.

5.1.1 Federal Laws and Regulations

National Environmental Policy Act of 1969, as amended (42 U.S.C. 4321 *et seq.*)—The National Environmental Policy Act (NEPA) establishes a national policy promoting awareness of the environmental consequences of human activity on the environment and consideration of environmental impacts during the planning and decision-making stages of a project. It requires Federal agencies to prepare a detailed EIS for any major Federal action that could have a significant environmental impact.

DOE has prepared this EIS in accordance with the requirements of NEPA or as implemented by Council on Environmental Quality regulations (40 CFR 1500 *et seq.*) and DOE NEPA implementing regulations (10 CFR 1021).

Atomic Energy Act of 1954 (42 U.S.C. 2011 *et seq.*)—The Atomic Energy Act authorizes DOE to establish standards to protect health or minimize dangers to life or property for activities under DOE's jurisdiction. Through a series of DOE Orders, an extensive system of standards and requirements has been established to ensure safe operation of facilities. DOE regulations generally are found in Title 10 of the Code of Federal Regulations (CFR).

The Nuclear Waste Policy Act of 1982 (U.S.C. 10101 through 10271)—The Nuclear Waste Policy Act established the Office of Civilian Radioactive Waste Management (42 U.S.C. 10224) and the Nuclear Waste Fund (42 U.S.C. 10222) and defined its mission to develop a Federal system for the management and geologic disposal of commercial spent nuclear fuel and other high-level radioactive wastes. As originally enacted, it

called for the Secretary of Energy to recommend candidate repository sites, but in 1987 it was amended to require DOE to proceed with characterization of the Yucca Mountain site only (42 U.S.C. 10133 and 10172). The Energy Policy Act of 1992, Section 801, directed the U.S. Environmental Protection Agency (EPA) to promulgate public health and safety standards for the protection of the public from releases from radioactive materials stored or disposed of in the proposed repository at the Yucca Mountain site.

Low-Level Radioactive Waste Policy Act of 1980, as amended (42 U.S.C. 2021 *et seq*.)—This Act amended the Atomic Energy Act to specify that the Federal Government is responsible for disposal of low-level radioactive waste generated by its activities, and the states are responsible for disposal of other low-level radioactive waste. It provides for and encourages interstate compacts to carry out the state responsibilities.

Solid Waste Disposal Act of 1965, as amended by the Resource Conservation and Recovery Act of 1976 and the Hazardous and Solid Waste Amendments of 1984 (42 U.S.C. 6901 *et seq*.)—The Solid Waste Disposal Act of 1965, as amended, governs the transportation, treatment, storage, and disposal of hazardous and nonhazardous waste. Under the Resource Conservation and Recovery Act of 1976 (RCRA), which amended the Solid Waste Disposal Act of 1965, the EPA defines and identifies hazardous waste; establishes standards for its transportation, treatment, storage, and disposal; and requires permits for persons engaged in hazardous waste activities. Section 3006 of the Act (42 U.S.C. 6926) allows states to establish and administer those permit programs with EPA approval. The EPA regulations implementing RCRA are found in 40 CFR Parts 260 through 283.

Regulations imposed on a generator or a treatment, storage, and/or disposal facility vary according to the type and quantity of material or waste generated, treated, stored, and/or disposed of. The method of treatment, storage, and/or disposal also impacts the extent and complexity of the requirements.

Federal Facilities Compliance Act of 1992 (42 U.S.C. 6961 *et seq*.)—Section 102(a)(3) of the Federal Facilities Compliance Act waives sovereign immunity for Federal facilities for fines and penalties for RCRA violations and state, interstate, and local hazardous and solid waste management requirements. This waiver was delayed for three years following enactment for violations of the land disposal restrictions storage and prohibition (RCRA section 3004(j)) involving mixed waste at DOE facilities. The Act further delays the waiver of sovereign immunity beyond the three-year period at a facility if DOE is in compliance with an approved plan for developing treatment capacity and technologies for mixed waste generated or stored at the facility, as well as with an Order requiring compliance with the plan.

DOE and the State of Idaho have an approved plan, known as the "Site Treatment Plan," and associated consent order. Some of the waste being analyzed in this EIS has been designated for treatment according to terms in the Idaho National Engineering and Environmental Laboratory (INEEL) Site Treatment Plan. If DOE makes a decision based on this EIS that differs from that agreed to with the State of Idaho in the Site Treatment Plan, that Plan would be subject to renegotiation.

Toxic Substances Control Act of 1976 (15 U.S.C. 2601 *et seq*.)—The Toxic Substances Control Act provides the EPA with the authority to require testing of chemical substances entering the environment and to regulate them as necessary. The law complements and expands existing toxic substance laws, such as Section 112 of the Clean Air Act and Section 307 of the Clean Water Act. The Toxic Substances Control Act requires compliance with inventory reporting and chemical control provisions of the Act to protect the public from the risks of exposure to chemicals. The Act also imposes strict limitations on the use and disposal of polychlorinated biphenyls, chlorofluorocarbons, asbestos, dioxins, certain metal-working fluids, and hexavalent chromium. Some disposal activities under this Act might require a permit from EPA.

Clean Air Act of 1970 (42 U.S.C. 7401 *et seq*.)—The Clean Air Act 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 (42 U.S.C. 7418) requires that each Federal agency with jurisdiction over any property or facility that might result in the discharge of air pollutants comply with "all Federal, state, interstate, and local requirements" with regard to the control and abatement of air pollution.

The Clean Air Act: (1) requires the EPA to establish National Ambient Air Quality Standards (NAAQS) as necessary to protect the public health, with an adequate margin of safety, from any known or anticipated adverse effects of a regulated pollutant (42 U.S.C. 7409 *et seq.*); (2) requires establishment of national standards of performance for new or modified stationary sources of atmospheric pollutants (42 U.S.C. 7411); (3) requires specific emission increases to be evaluated so as to prevent a significant deterioration in air quality (42 U.S.C. 7470 *et seq.*); and (4) requires specific standards for releases of hazardous air pollutants (including radionuclides) (42 U.S.C. 7412). These standards are implemented through state implementation plans developed by each state with EPA approval. The Clean Air Act requires sources to meet standards and obtain permits to satisfy these standards.

Hazardous air pollutants are substances that may cause health and environmental effects at low concentrations. Currently, 189 compounds have been identified as hazardous air pollutants. A major source is defined as any stationary source, or a group of stationary sources located within a contiguous area under common control, that emits or has the potential to emit at least 10 tons per year of any single hazardous air pollutant or 25 tons per year of a combination of pollutants.

The 1990 amendments to the Clean Air Act substantially revised the program to regulate potential emissions of hazardous air pollutants. The aim of the new control program is to require state-of-the-art pollution control technology on most existing and all new emission sources. These provisions regulate emissions by promulgating emissions limits reflecting use of the maximum achievable control technology. These emission limits are then incorporated into a facility's operating permit. Air emissions are regulated by the EPA under 40 CFR Parts 50 through 99.

Radionuclide emissions other than radon from DOE facilities are also covered under the National Emission Standards for Hazardous Air Pollutants program (40 CFR 61.90-97). To determine compliance with the standard, an effective dose equivalent value for the maximally exposed members of the public is calculated using EPA-approved sampling procedures, computer models, or other EPA-approved procedures. DOE is currently determining if a National Emission Standards for Hazardous Air Pollutants permit will be required for radiological emissions from any spent nuclear fuel treatment and management facilities at the Savannah River Site (SRS) (stacks, process vents, etc.).

Clean Water Act of 1972 (33 U.S.C. 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 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.

The Clean Water Act provides water quality standards for the Nation's waterways, guidelines and limitations for effluent discharges from point-source discharges, and the National Pollutant Discharge Elimination System (NPDES) permit program. The NPDES program is administered by the Water Management Division of the EPA pursuant to regulations in 40 CFR Part 122 *et seq*. Sections 401 through 405 of the Water Quality Act of 1987 added Section 402(p) to the Clean Water Act to require the EPA to establish regulations for permits for stormwater discharges associated with industrial activities. Stormwater provisions of the NPDES program are set forth at 40 CFR 122.26. Permit modifications are required if the discharge effluent is altered. DOE will apply for discharge permit for spent nuclear fuel treatment and management facilities at SRS if the treatment process results in discharges to waters of South Carolina.

Safe Drinking Water Act of 1974, as amended (42 U.S.C. 300(f) *et seq*.)—The primary objective of the Safe Drinking Water Act is to protect the quality of public drinking water supplies and sources of drinking water. The implementing regulations, administered by the EPA unless delegated to the states, establish standards applicable to public water systems. These regulations include maximum contaminant levels (including those for radioactivity) in public water systems, which are defined as water systems that have at least 15 service connections used by year-round residents or regularly serve at least 25 year-round residents. The EPA regulations implementing the Safe Drinking Water Act are found under 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 4 millirem per year beta activity (40 CFR 141.16(a)). 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.

The States of Idaho and South Carolina have received authorization from EPA to implement the public drinking water system program and the underground injection control program under the Safe Drinking Water Act. The Division of Environmental Quality, as a subdivision of the Idaho Department of Health and Welfare, sets forth monitoring and reporting requirements for inorganic and organic chemicals, and radiochemicals in Idaho. The South Carolina Department of Health and Environmental Control has established similar requirements for South Carolina.

The Safe Drinking Water Act also provides for designation of aquifers to be protected from degradation due to their importance as the sole source of drinking water. The Snake River Plain aquifer underlying INEEL has been designated as a sole source aquifer by EPA (40 FR 100-109, October 7, 1991) because groundwater supplies 100 percent of the drinking water consumed within the Eastern Snake River Plain and an alternative source or sources is not available.

Hazardous Material Transportation Act of 1975 (49 U.S.C. 5105 *et seq*.)—The Hazardous Material Transportation Act requires the U.S. Department of Transportation to prescribe uniform national regulations for transportation of hazardous materials (including radioactive materials). Most state and local regulations regarding such transportation that are not substantively the same as the Department of Transportation regulations are preempted (i.e., rendered void) (49 U.S.C. 5125). This, in effect, allows state and local governments only to enforce the Federal regulations, not to change or expand upon them.

This program is administered by the Research and Special Programs Administration of the Department of Transportation, which coordinates its regulations with those of the U.S. Nuclear Regulatory Commission (NRC), under the Atomic Energy Act, and with the EPA, under RCRA, when covering the same activities.

Individual states and Tribes often have their own statutes and/or regulations governing transportation of hazardous or radioactive materials. These laws might also be applicable to DOE transportation activities. An example of a local law that affects transportation of materials offsite from the INEEL is the Shoshone-Bannock Tribal Ordinance, the Nuclear Materials Transportation Act, ENVR 92-S5, which restricts transportation of radioactive materials across the Shoshone-Bannock Reservation.

National Historic Preservation Act of 1966, as amended (16 U.S.C. 470 *et seq*.)—The National Historic Preservation Act provides that sites with significant national historic value be placed on the *National Register of Historic Places*, which is maintained by the Secretary of the Interior. Section 110 of the Act requires Federal agencies to identify, evaluate, inventory, and protect National Register resources on properties under their control. No permits or certifications are required under the Act. However, if a particular Federal activity may impact a historic property resource, consultation with the Advisory Council on Historic Preservation is required under 16 U.S.C. 470(f). Such consultation usually generates a Memorandum of Agreement, including stipulations that must be followed to minimize adverse impacts.

Coordination with the state Historic Preservation Officer also is undertaken to ensure that potentially significant sites are identified properly and appropriate mitigative actions are implemented. DOE has notified respective State Historic Preservation Offices of its intent to consult on this project.

Endangered Species Act of 1973 (16 U.S.C. 1531 *et seq*.)—The Endangered Species Act is intended to prevent the further decline of endangered and threatened species and to restore these species and habitats. Section 7 of the Act requires Federal agencies that have reason to believe a prospective action may affect an endangered or threatened species or its habitat to consult with the U.S. Department of the Interior to ensure that the action does not jeopardize the species or destroy its habitat. If, despite reasonable and prudent measures to avoid or minimize such impacts, the species or its habitat would be jeopardized by the action, a review process is specified to determine whether the action may proceed. DOE has consulted with the U.S. Fish and Wildlife Service regarding impacts on any species listed under the Endangered Species Act.

American Indian Religious Freedom Act of 1978 (42 U.S.C. 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 religions.

Occupational Safety and Health Act of 1970 (29 U.S.C. 651 *et seq*.)—The Occupational Safety and Health Act establishes standards for 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 (OSHA), a U.S. Department of Labor agency. Although OSHA and the EPA both have a mandate to reduce exposures to toxic substances, OSHA's jurisdiction is limited to safety and health conditions that exist in the workplace environment.

Under the Act, it is the duty of each employer to furnish employees a place of employment free of recognized hazards that are likely to cause death or serious physical harm. Employees have a duty to comply with the occupational safety and health standards and rules, regulations, and Orders issued under the Act. OSHA regulations (29 CFR) establish specific standards that tell employers what must be done to achieve a safe and healthful working environment. Government agencies, including DOE, are not technically subject to OSHA regulations, but are required under 29 U.S.C. 668 to establish their own occupational safety and health programs for their places of employment which are consistent with OSHA standards. 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 Order 5480.1B and 54831.A). DOE keeps and makes available the various records of minor illnesses, injuries, and work-related deaths as required by OSHA regulations.

Pollution Prevention Act of 1990 (42 U.S.C. 13101 *et seq*.)—The Pollution Prevention Act establishes a national policy for waste management and pollution control. Source reduction is given first preference, followed by environmentally safe recycling, with disposal or releases to the environment as a last resort. In response to the policies established by the Act, DOE committed to participation in the Superfund Amendments and Reauthorization Act, Section 313, EPA 33/50 Pollution Prevention Program. The goal for facilities involved in compliance with Section 313 was to achieve a 33 percent reduction (from a 1993 baseline) in the release of 17 priority chemicals by 1997. On August 3, 1993, President Clinton issued Executive Order 12856, which required DOE to achieve a 50 percent reduction in total releases of all toxic chemicals by December 31, 1999.

Noise Control Act of 1972, as amended (42 U.S.C. 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.

5.1.2 Executive Orders

Executive Order 11514 (Protection and Enhancement of Environmental Quality)—Executive Order 11514 requires Federal agencies to continually monitor and control their activities 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 the Federal plans and programs with environmental impact to obtain the views of interested parties. DOE has issued regulations (10 CFR 1021) and DOE Order 5440.1E for compliance with this Executive Order.

Executive Order 11593 (National Historic Preservation, May 13, 1971)—Executive Order 11593 directs Federal agencies to locate, inventory, and nominate properties under their jurisdiction or control to the *National Register of Historic Places* if those properties qualify. This process requires DOE to provide the Advisory Council on Historic Preservation the opportunity to comment on the possible impacts of the proposed activity on any potential eligible or listed resources.

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. DOE requirements for compliance with flood plain and wetlands activity are codified in 10 CFR 1022.

Executive Order 12088 (Federal Compliance with Pollution Control Standards, October 13, 1978, as amended by Executive Order 12580, Federal Compliance with Pollution Control Standards, January 23, 1987)—Executive Order 12088 directs Federal agencies to comply with applicable administrative and procedural pollution control standards established by, but not limited to, the Clean Air Act, the Noise Control Act, the Clean Water Act, the Safe Drinking Water Act, the Toxic Substances Control Act, and RCRA.

Executive Order 12580 (Superfund Implementation)—Executive Order 12580 delegates to the heads of executive departments and agencies the responsibility for undertaking: (1) remedial actions for releases or threatened releases that are not on the National Priority List, and (2) removal actions, other than emergencies, where the release is from any facility under the jurisdiction or control of executive departments and agencies.

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 any disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority or low-income populations.

Executive Order 13101 (Greening the Government Through Waste Prevention, Recycling, and Federal Acquisition)—Executive Order 13101 requires Federal agencies to incorporate waste prevention and recycling

in its daily operations and work to increase and expand markets for recovered materials. This Order states that it is national policy to prefer pollution prevention, whenever feasible. Pollution that cannot be prevented should be recycled; pollution that cannot be prevented or recycled should be treated in an environmentally safe manner. Disposal should be employed only as a last resort.

5.1.3DOE Directives

The Atomic Energy Act authorizes DOE to establish standards to protect health or minimize dangers to life or property from activities under DOE's jurisdiction. Through a series of DOE Directives, an extensive system of policies, Orders, notices, manuals, and guides has been established to ensure safe operation of facilities.

DOE regulations are generally found in Title 10 of the Code of Federal Regulations. These regulations address such areas as energy conservation, administrative requirements and procedures, nuclear safety, and classified information. For the purpose of this EIS, relevant regulations include: Procedural Rules for DOE Nuclear Activities (10 CFR 820); Nuclear Safety Management (10 CFR 830); Radiation Protection of the Public and the Environment (10 CFR 834, Draft); Occupational Radiation Protection (10 CFR 835); Compliance with the National Environmental Policy Act (10 CFR 1021); and Compliance with Floodplains/Wetlands Environmental Review Requirements (10 CFR 1022).

DOE Directives are issued in support of health, safety, and environmental programs. Many of DOE's Directives are in the process of being revised and reorganized to reduce duplication and eliminate obsolete provisions. The new DOE Directives are organized by series, with each Directive identified by a letter and three digit number, and will include all DOE policies, Orders, notices, manuals, and guides. The remaining DOE Directives are expected to be revised and converted to the new DOE numbering system over the next two years. The major DOE Directives pertaining to the proposed action and alternatives are listed in **Table 5–1**.

5.1.4State Environmental Laws, Regulations, and Agreements

Certain environmental requirements have been delegated to state authorities for implementation and enforcement. It is DOE policy to conduct its operations in an environmentally safe manner in compliance with all applicable laws, regulations, and standards, including state laws and regulations. A list of potentially applicable state laws, regulations, and agreements are provided in **Table 5–2**.

5.2 RADIOACTIVE MATERIAL PACKAGING AND TRANSPORTATION REGULATIONS

Transportation of hazardous and radioactive materials and substances are governed by the Department of Transportation and the NRC. Department of Transportation regulations, which may be found under 49 CFR Parts 171 through 178 and 49 CFR Parts 383 through 397, contain requirements for identifying a material as hazardous or radioactive. These regulations interface with NRC regulations for identifying material, but the Department of Transportation hazardous material regulations govern the hazard communication (such as marking, hazard labeling, vehicle placarding, and emergency response telephone number) and shipping requirements.

The NRC regulations applicable to radioactive materials transportation may be found under 10 CFR Part 71. These regulations include detailed packaging design requirements and package certification testing requirements. Complete documentation of design and safety analysis and results of the required testing are submitted to the NRC to certify the package for use. This certification testing involves the following components: heat, physical drop onto an unyielding surface, water submersion, puncture by dropping package onto a steel bar, and gas tightness. DOE may certify its own packages, per 49 CFR 173.7(d).

Table 5–1 Relevant DOE Directives

DOE Directive	*Subject*
Leadership/Management Planning	
O 151.1	Comprehensive Emergency Management System (09/25/95; Change 2, 08/21/96)
Information and Analysis	
O 231.1	Environment, Safety, and Health Reporting (09/30/95; Change 2, 11/07/96)
O 232.1A	Occurrence Reporting and Processing of Operations Information (07/21/97)
Work Processes	
O 414.1A	Quality Assurance (9/29/99)
O 420.1	Facility Safety (10/13/95; Change 2, 10/24/96)
O 435.1	Radioactive Waste Management (07/09/99)
O 440.1A	Worker Protection Management for DOE Federal and Contractor Employees (03/27/98)
N 441.1	Radiological Protection for DOE Activities (09/30/95, extended until 06/30/00 by N 441.4, 11/20/98)
N 441.4	Extension of DOE N 441.1, Radiological Protection for DOE Activities (11/20/98)
O 451.1A	National Environmental Policy Act Compliance Program (06/05/97)
O 460.1A	Packaging and Transportation Safety (10/02/96)
O 460.2	Departmental Materials Transportation and Packaging Management (09/27/95; Change 1, 10/26/95)
O 470.1	Safeguards and Security Program (09/28/95; Change 1, 06/21/96)
O 470.2	Safeguards and Security Independent Oversight Program (12/23/98)
O 474.1	Control and Accountability of Nuclear Materials (8/11/99)
Personnel Relations and Services	
3790.1B	Federal Employee Occupational Safety and Health Program (01/07/93)
Real Property Management	
4330.4B	Maintenance Management Program (02/10/94)
Project Management	
4700.1	Project Management System (03/06/87; Change 1, 06/02/92)
Environmental Quality and Impact	
5400.1	General Environmental Protection Program (11/09/88; Change 1, 06/29/90)
5400.5	Radiation Protection of the Public and the Environment (02/08/90; Change 2, 01/07/93)
5480.4	Environmental Protection, Safety, and Health Protection Standards (05/15/84; Change 4, 01/07/93)
5480.19	Conduct of Operations Requirements for DOE Facilities (07/09/90; Change 1, 05/18/92)
5480.20A	Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities (11/15/94)
5480.21	Unreviewed Safety Questions (12/24/91)
5480.22	Technical Safety Requirements (02/25/92; Change 2, 01/23/96)
5480.23	Nuclear Safety Analysis Report (04/10/92; Change 1, 03/10/94)
5480.30	Nuclear Reactor Safety Design Criteria (01/19/93)
5484.1	Environmental Protection, Safety, and Health Protection Information Reporting Requirements (02/24/81; Change 7, 10/17/90)
Emergency Preparedness	
5530.3	Radiological Assistance Program (01/14/92; Change 1, 04/10/92)
5530.5	Federal Radiological Monitoring and Assessment Center (07/10/92; Change 1 12/02/92)
Defense Programs	
5610.14	Transportation Safeguards System Program Operations (05/12/93)
5632.1C	Protection and Control of Safeguards and Security Interests (07/15/94)
5632.7A	Protective Force Program (04/13/94; Change 1, 02/13/95)
5660.1B	Management of Nuclear Materials (05/26/94)
Design	
6430.1A	General Design Criteria (04/06/89)

Table 5–2 State Environmental Laws, Regulations, and Agreements

Law/Regulation/Agreement	*Citation*	*Potential Requirements*
Idaho National Engineering Environmental Laboratory (INEEL), Idaho		
Idaho Environmental Protection and Health Act	ID Code, Title 39, Chapter 1	Provides for development of air pollution control permitting regulations.
Idaho Air Pollution Control Act	ID Code, Title 39, Chapter 29	Requires permitting of sources and control of toxic air pollutants and other pollutants.
Rules for the Control of Air Pollution in Idaho	IDAPA 16, Title 01, Chapter 01	Enforces national ambient air quality standards.
Idaho Water Pollution Control Act	ID Code, Title 39, Chapter 36	Enhances and preserves the quality and the value of water resources.
Idaho Rules for Public Drinking Water Systems	IDAPA 16, Title 01, Chapter 08	Controls and regulates the design, construction, operation, maintenance, and quality control of public drinking water.
Water Quality Standards and Wastewater Treatment Regulations	IDAPA 16, Title 01, Chapter 02	Enforces standards relating to the discharge of effluent into the water.
Transportation of Hazardous Waste	ID Code, Title 18, Chapter 39 ID Code, Title 49, Chapter 22	Regulates transportation of hazardous materials/hazardous waste on highways.
Various Acts Regarding Fish and Game	ID Code, Title 36, Chapters 9, 16 and 19	Requires consultation with responsible agency.
Endangered Species Act	ID Code, Title 67, Chapter 8	Requires consultation with Department of Fish and Game.
Classification and Protection of Wildlife	IDAPA 13, Title 01, Chapter 06	Requires consultation with Department of Fish and Game.
Idaho Historic Preservation	ID Code, Title 67, Chapters 41 and 46	Requires consultation with responsible local governing body.
Memorandum of Agreement	January 26, 1994	Requires consultation with Shoshone-Bannock Tribes.
Agreement-in-Principal (formerly Tribal Working Agreement)	August 6, 1998	Establishes understanding and commitment between the Tribes and DOE.
Federal Facility Agreement and Consent Order	December 9, 1991	Establishes a process for evaluating past potential releases to the environment at Idaho National Engineering and Environmental Laboratory (INEEL).
Spent Fuel Settlement Agreement (also known as the Batt Agreement)	October 16, 1995	Allows INEEL to receive spent nuclear fuel and mixed waste from off site and establishes schedules for the treatment of high-level radioactive waste, removal of spent nuclear fuel from the state, and treatment of mixed waste.

Law/Regulation/Agreement	*Citation*	*Potential Requirements*
Savannah River Site, South Carolina		
South Carolina Pollution Control Act	SC Code, Title 48, Chapter 1	Provides for the development of air pollution permitting regulations and air pollution control regulations
South Carolina Air Pollution Control Regulations and Standards	R.61-62	Requires permit prior to construction or modification of an air contaminant source and control of toxic air pollutants and other pollutants.
South Carolina Atomic Energy & Radiation Control Act	SC Code, Title 13, Chapter 7	Establishes standards for radioactive air emissions.
South Carolina Atomic Energy & Radiation Regulations and Standards	R.61-63 R.61-83	Establishes standards for radioactive air emissions.
South Carolina Pollution Control Act-Water	SC Code, Title 48, Chapter 1	Requires permit prior to construction or modification of a water discharge source.
South Carolina Water Pollution Control Regulations and Standards	R.61-9	Requires permit for the discharge of pollutants from any point source into waters of the state.
South Carolina Water Classification and Standards	R.61-68	Establishes official classified water uses, rules, and specific numeric water quality standards for protecting classified and existing water uses.
South Carolina Safe Drinking Water Act	SC Code, Title 44, Chapter 55	Establishes drinking water standards.
South Carolina Hazardous Waste Regulations and Standards	R.61-79 R.61-99 R.61-104	Protects human health and the environment by requiring careful management practices of hazardous waste.
South Carolina Solid Waste and Policy Management Act	SC Code, Title 44, Chapter 96	Establishes standards to treat, store, or dispose of solid waste.
South Carolina Solid Waste Regulations and Standards	R.61-107	Requires permit to store, collect, dispose, or transport solid waste.
South Carolina Nongame and Endangered Species Conservation Act	SC Code, Title 50, Chapter 15	Requires consultation with Wildlife and Marine Resources Department and minimization of impact.
South Carolina Museum Commission and Archaeology and Anthropology	Title 60, Chapter 12	Requires consultation with state Historic Preservation Office and minimization of impact.

Transportation casks, which are used to transport the radioactive material, are subject to numerous inspections and tests (10 CFR 71.87). These tests are designed to ensure that the cask components are properly assembled and meet applicable safety requirements. Tests and inspections are identified clearly in the Safety Analysis Report for Packaging and/or the Certificate of Compliance for each cask. Casks are loaded and inspected by registered users in compliance with approved quality assurance programs. Operations involving the casks are conducted in compliance with 10 CFR 71.91. Reports of defects or accidental mishandling are submitted to the NRC.

5.3 EMERGENCY MANAGEMENT AND RESPONSE LAWS, REGULATIONS, AND EXECUTIVE ORDERS

This section discusses the laws, regulations, and Executive Orders applicable to emergency management and response for the proposed action and alternatives.

5.3.1 Federal Laws

Emergency Planning and Community Right-to-Know Act of 1986 (U.S.C. 11001 *et seq*.) (also known as "SARA Title III")—This Act requires emergency planning and notice to communities and government agencies of the presence and release of specific chemicals. The EPA implements this Act under regulations found at 40 CFR Parts 355, 370, and 372. Under Subtitle A of this Act, Federal facilities are required to provide various information (such as inventories of specific chemicals used or stored and releases that occur from these sites) to the state emergency response commission and to 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 emission reporting began in 1988. DOE requires compliance with Title III as a matter of DOE policy.

Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (42 U.S.C. 9604(I) (also know as "Superfund")—This Act provides authority for Federal and state governments to respond directly to hazardous substances incidents. The Act requires reporting of spills, including radioactive spills, to the National Response Center.

Robert T. Stafford Disaster Relief and Emergency Assistance Act of 1988 (42 U.S.C. 5121)—This Act, 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 this Act. The President, in Executive Order 12148, delegated all functions except those in Section 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.

Justice Assistance Act of 1984 (42 U.S.C. 3701-3799)—This Act establishes Emergency Federal Law Enforcement Assistance to assist state and local governments in responding to a law enforcement emergency. The Act defines the term "law enforcement emergency" as an uncommon situation that requires law enforcement and is or threatens to become serious or of epidemic proportions, 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 (e.g., the Mount Saint Helens volcanic eruption) are eligible for Federal law enforcement assistance that includes funds, equipment, training, intelligence information, and personnel.

5.3.2 Federal Regulations

Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release (10 CFR 30.72, Schedule C)—This section of the regulations provides a list that is the basis used by both the public and private sector to determine if the radiological materials they handle must have an emergency response plan for unscheduled releases, and is one of the threshold criteria documents for DOE Hazards Assessments required by DOE Order 5500.3A, *Planning and Preparedness for Operational Emergencies*. The Federal Radiological Emergency Response Plan, dated November 1995, primarily discusses offsite Federal response in support of state and local governments with jurisdiction during a peacetime radiological emergency.

Occupational Safety and Health Administration Emergency Response, Hazardous Waste Operations, and Worker Right to Know (29 CFR)—This regulation establishes the OSHA 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 1920.120), and hazards

communication (Section 1910.1200) that enable 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 and Communications, Emergency Response Information Requirements (49 CFR 172)—This regulation defines the regulatory requirements for marking, labeling, placarding, and documenting hazardous materials shipments. The regulation also specifies the requirements for providing hazardous material information and training.

5.3.3 Executive Orders

Executive Order 12148 (Federal Emergency Management, July 20, 1979)—Executive Order 12148 transfers functions and responsibilities associated with Federal emergency management to the Director of the Federal Emergency Management Agency. The Order assigns the Director 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 12656 (Assignment of Emergency Preparedness Responsibilities, November 1988)—Executive Order 12656 assigns emergency preparedness responsibilities to Federal departments and agencies.

5.4 CONSULTATIONS

Certain laws, such as the Endangered Species Act, the National Historic Preservation Act, and the American Indian Religious Freedom Act, recommended that consultation and coordination with other Federal agencies, state and local agencies, and Federally recognized Native American groups take place prior to a prospective action to ensure that the action does not jeopardize or destroy important resources. These consultations must occur on a timely basis before any proposed action can begin.

Consultations associated with the proposed action involve biotic resources, cultural resources, and Native American religious rights. Biotic resources consultations are to address the potential for the proposed action to disturb sensitive species or habitats. Cultural resources consultations are to address the potential disruption of important cultural resources and archaeological sites. Native American consultations are to address any potential disturbance of ancestral Native American sacred sites and traditional resources and practices. DOE consulted with the appropriate agencies, as discussed in Chapters 3 and 4.

6. GLOSSARY

Air Pollutant — Any substance in the air which could, if in a high-enough concentration, harm man, animals, vegetation, or material.

Air Quality Control Region — A geographic subdivision of the United States, designed to deal with pollution on a regional or local level. Some regions span more than one state.

Alluvial Deposits — Deposits of earth, sand, gravel, and other materials carried by moving surface water and deposited at points of weak water flow.

Alpha Particle — A positively charged particle, consisting of two protons and two neutrons, that is emitted during radioactive decay from the nucleus of certain nuclides. It is the least penetrating of the three common types of radiation (alpha, beta, and gamma).

Alpha Waste — Waste containing radioactive isotopes that decay by producing alpha particles.

Ambient Air — The surrounding atmosphere as it exists around people, plants, and structures. Air quality standards are used to provide a measure of the health-related and visual characteristics of the air.

Ambient Air Quality Standards — The level of pollutants in the air prescribed by regulations that may not be exceeded during a specified time in a defined area.

Aquatic — Living or growing in, on, or near water.

Aquifer — A saturated geologic unit through which significant quantities of water can migrate under natural hydraulic gradients.

Archaeological Resources — Any location where humans have altered the terrain or discarded artifacts during either prehistoric or historic times.

Artifact — An object produced or shaped by human workmanship of archaeological or historical interest.

As Low as Reasonably Achievable — A concept applied to ensure the quantity of radioactivity released to the environment and the radiation exposure of onsite workers in routine operations, including "anticipated operational occurrences," is maintained as low as reasonably achievable. It takes into account the state of technology, economics of improvements in relation to benefits to public health and safety, and other societal and economic considerations in relation to the use of nuclear energy in the public interest.

Background Radiation — Ionizing radiation present in the environment from cosmic rays and natural sources in the Earth; background radiation varies considerably with location. The U.S. average background radiation is 300 millirem per year.

Badged Worker — A worker who has the potential to be exposed to radiation and is equipped with a dosimeter to measure his/her dose.

Barrier — Any material or structure that prevents or substantially delays movement of radionuclides toward the accessible environment.

Baseline — A quantitative expression of conditions, costs, schedule, or technical progress to serve as a base or standard for measurement during the performance of an effort; the established plan against which the status of resources and progress of a project can be measured. For this environmental impact statement, the environmental baseline is the site environmental conditions as they exist or have been estimated to exist in the absence of the proposed action.

BEIR V — Biological Effects of Ionizing Radiation; referring to the fifth in a series of committee reports from the National Research Council.

Beta Particle — A charged particle emitted from the nucleus of an atom during radioactive decay. A negatively charged beta particle is identical to an electron; a positively charged beta particle is called a "positron."

Biota (biotic) — The plant and animal life of a region (pertaining to biota).

Blanket Fuel — Those fuel tubes or elements composed of depleted or natural enrichment of uranium, placed at the perimeter of the reactor core, and used to breed the fissile material plutonium-239 or used as shielding.

Blending — Selecting spent nuclear fuel assemblies of different characteristics for inclusion in a transportation cask, storage mode, or waste package; also, selecting high-level radioactive waste of different characteristics for inclusion in a transportation cask or waste package to meet design goals.

Borosilicate Waste Glass — A glass that typically contains approximately 20 to 40 weight percent waste oxides, 40 to 65 weight percent silica, 5 to 10 weight percent boron oxide, and 10 to 20 weight percent alkali oxides, plus other oxide constituents.

Breeder Reactor — A type of nuclear reactor that creates more fissionable fuel than it uses.

Burnup — A term used to indicate the amount of fuel consumed during the irradiation process. The percentage of heavy metal atoms fissioned or the thermal energy produced per mass of fuel (usually measured in megawatt days per ton [MWd/t]).

Calcine — To heat to a high temperature without fusing in order to decompose or oxidize; the material produced by converting high-level radioactive waste to unconsolidated granules or powder.

Cancer — The name given to a group of diseases characterized by uncontrolled cellular growth with cells having invasive characteristics such that the disease can transfer from one organ to another.

Canister — The structure surrounding the waste form (e.g., high-level radioactive waste immobilized in borosilicate glass) that facilitates handling, storage, transportation, and/or disposal. A canister is a metal receptacle with the following purpose: (1) for solidified high-level radioactive waste, it is used as a pour mold and (2) for spent nuclear fuel, it may provide structural support for intact spent nuclear fuel, loose rods, nonfuel components, or confinement of radionuclides.

Canning — The process of placing spent nuclear fuel in canisters to retard corrosion, contain radioactive releases, or control geometry.

Capable Fault — A fault that has exhibited one or more of the following characteristics:

(1) Movement at or near the ground surface at least once within the past 35,000 years or movement of a recurring nature within the past 500,000 years.

(2) Macro-seismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault.

(3) A structural relationship to a capable fault according to characteristics (1) or (2) of this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other.

Cask — A heavily shielded container that meets U.S. Nuclear Regulatory Commission and U.S. Department of Transportation regulatory requirements and is used to store and/or ship radioactive materials (i.e., spent nuclear fuel or high-level radioactive waste). Lead, depleted uranium, and steel are common materials used in the manufacture of casks.

Characterization — The determination of waste composition and properties, whether by review of process knowledge, nondestructive examination or assay, or sampling and analysis, generally done for the purpose of determining appropriate storage, treatment, handling, transport, and disposal requirements.

Chronic Exposure — Low-level radiation exposure incurred over a long time period due to residual contamination.

Cladding — The outer jacket of a fuel element, cladding is usually made of aluminum, stainless steel, or zirconium alloy. It is used to prevent fuel corrosion and retain fission products during reactor operation or to prevent releases into the environment during storage.

Class I Areas — National parks and wilderness areas designated by the Prevention of Significant Deterioration section of the Clean Air Act amendments. These amendments and the implementing regulations provide special protection to air quality and air quality-related values in such areas. Only very slight deterioration of air quality is allowed in Class I areas.

Class II Areas — Most of the country not designated as Class I is designated as Class II. Class II areas are generally cleaner than air quality standards require and moderate increases in new pollution are allowed after a regulatory-mandated impacts review.

Code of Federal Regulations (CFR) — All Federal regulations in force are published in codified form in the Code of Federal Regulations.

Collective Committed Effective Dose Equivalent — The committed effective dose equivalent of radiation for a population.

Committed Dose Equivalent — The predicted total dose equivalent to a tissue or organ over a 50-year period after an intake of a radionuclide into the body. It does not include external dose contributions. Committed dose equivalent is expressed in units of rem or sievert. The committed effective dose equivalent is the sum of the committed dose equivalents to various tissues of the body, each multiplied by the appropriate weighting factor.

Community (biotic) — All plants and animals occupying a specific area under relatively similar conditions.

Conditioning — Any process which prepares or treats spent nuclear fuel or high-level radioactive waste for storage, transportation, or disposal in accordance with regulatory requirements (e.g., melt and dilute product).

Conformity — Conformity is defined in the Clean Air Act as the action's compliance with an implementation plan's purpose of eliminating or reducing the severity and number of violations of the National Ambient Air Quality Standards and achieving expeditious attainment of such standards; and that such activities will not: (1) cause or contribute to any new violation of any standard in any area; (2) increase the frequency or severity of any existing violation of any standard in any area; or (3) delay timely attainment of any standard or any required interim emission reduction or other milestones in any area.

Consumptive Water Use — The difference in the volume of water withdrawn from a body of water and the amount released back into the body of water.

Colluvial Deposits — Loose deposits of rock and other material, usually at the foot of a slope or cliff and brought there chiefly by gravity.

Contact-handled Waste — Packaged waste whose external surface dose rates does not exceed 200 millirem per hour.

Container — With regard to radioactive waste, the metal envelope in the waste package that provides the primary containment function of the waste package and is designed to meet the containment requirements of 10 CFR 60.

Contamination — The deposition of unwanted radioactive material on the surfaces of structures, areas, objects, or personnel.

Coolant — A gas or liquid circulated through a nuclear reactor to remove or transfer heat.

Credible Accident — An accident that has a probability of occurrence greater than or equal to one in a million years.

Criteria Pollutants — The Clean Air Act required the U.S. Environmental Protection Agency to set air quality standards for common and widespread pollutants after preparing "criteria documents" summarizing scientific knowledge on their health effects. Today there are standards in effect for six "criteria pollutants": sulfur dioxide (SO_2), carbon monoxide (CO), particulate matter less than or equal to 10 microns in diameter (PM_{10}) and less than or equal to 2.5 microns in diameter ($PM_{2.5}$), nitrogen dioxide (NO_2), ozone (O_3), and lead (Pb).

Critical Habitat — Defined in the *Endangered Species Act* of 1973 as "specific areas within the geographical area occupied by [an endangered or threatened] species, essential to the conservation of the species and which may require special management considerations or protection; and specific areas outside the geographical area occupied by the species that are essential for the conservation of the species."

Criticality — A self-sustained nuclear chain reaction resulting from fissionable material of sufficient mass in a particular geometry.

Cultural Resources — Archaeological sites, historical sites, architectural features, traditional use areas, and Native American sacred sites.

Cumulative Impacts — In an environmental impact statement, the impact on the environment which results from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions, regardless of what agency (Federal or nonfederal), private industry, or individual(s) undertakes such other actions. Cumulative impacts can result from individually minor but collectively significant actions taking place over a period of time (40 CFR 1508.7).

Curie (Ci) — A unit of radioactivity equal to 37 billion disintegrations per second; also a quantity of any nuclide or mixture of nuclides having 1 curie radioactivity.

Day-Night Average Sound Level — The 24-hour A-weighted (see decibel, A-weighted) equivalent sound level expressed in decibels, with a 10-decibel penalty added to sound levels between 10:00 p.m. and 7:00 a.m. to account for increased annoyance due to noise during nighttime hours.

Decay Heat (radioactivity) — The heat produced by the decay of certain radionuclides.

Decay (radioactive) — The decrease in the amount of any radioactive material with the passage of time due to the spontaneous transformation of an unstable nuclide into a different nuclide or into a different energy state of the same nuclide; the emission of nuclear radiation (alpha, beta, or gamma radiation) is part of the process.

Decladding — The process of mechanically removing the cladding from the fuel pin in a fuel element.

Decibel (dB) — A logarithmic unit of sound measurement which describes the magnitude of a particular quantity of sound pressure power with respect to a standard reference value. In general, a sound doubles in loudness for every increase of 10 decibels.

Decibel, A-weighted (dBA) — A unit of frequency weighted sound pressure level, measured by the use of a metering characteristic and the "A" weighting specified by the American National Standards Institution ANSI S1.4-1983 (R1594), that accounts for the frequency response of the human ear.

Deciduous — Trees which shed leaves at a certain season.

Decommissioning — The process of removing a facility from operation, followed by decontamination, entombment, dismantlement, or conversion to another use.

Decontamination — The actions taken to reduce or remove substances that pose a substantial present or potential hazard to human health or the environment, such as radioactive or chemical contamination from facilities, equipment, or soils by washing, heating, chemical or electrochemical action, mechanical cleaning, or other techniques.

Degraded (spent nuclear fuel) — Spent nuclear fuel whose external cladding has cracked, pitted, corroded, or potentially allows the leakage of radioactive materials.

°C (degrees Celsius) — A unit for measuring temperature using the centigrade scale in which the freezing point of water is 0 degrees and the boiling point is 100 degrees.

°F (degrees Fahrenheit) — A unit for measuring temperature using the Fahrenheit scale in which the freezing point of water is 32 degrees and the boiling point is 212 degrees.

Depleted Uranium — Uranium with a smaller percentage of uranium-235 than the 0.711 weight percent found in natural uranium. It is a byproduct of the uranium enrichment process, during which uranium-235 is collected from one batch of uranium, thereby depleting it, and adding to another batch to increase its concentration of uranium-235.

Dilute — To reduce the concentration of a substance by adding another material to it.

Disposal — The isolation of radioactive waste from the accessible environment, as defined in 10 CFR 60.2. Disposal means the emplacement in a repository of high-level radioactive waste, spent nuclear fuel, or other highly radioactive material with no foreseeable intent of recovery, whether or not such emplacement permits the recovery of such waste.

Direct Jobs — The number of workers required at a site to implement an alternative.

Disassembly — Removal of the fuel elements from the fuel assembly.

DOE Orders — Requirements internal to the U.S. Department of Energy (DOE) that establish DOE policy and procedures, including those for compliance with applicable laws.

DOE Site Boundary — A geographic boundary within which public access is controlled and activities are governed by the U.S. Department of Energy (DOE) and its contractors, not by local authorities. Based on the definition of exclusion zone, a public road traversing a DOE site is considered to be within the DOE site boundary if DOE or the site contractor has the capability to control the road at any time necessary.

Dose — The energy imparted to matter by ionizing radiation. The unit of absorbed dose is the rad.

Dose Commitment — The dose an organ or tissue would receive during a specified period of time (e.g., 50 to 100 years) as a result of intake (by ingestion or inhalation) of one or more radionuclides from a defined release, frequently over a year's time.

Dose Equivalent — The product of absorbed dose in rad (or Gray) and a quality factor, which quantifies the effect of this type of radiation in tissue. Dose equivalent is expressed in units of rem or Sievert, where 1 rem equals 0.01 Sievert.

Dosimeter — A small device (instrument) carried by a radiation worker that measures cumulative radiation dose (e.g., film badge or ionization chamber).

Drinking Water Standards — The level of constituents or characteristics in a drinking water supply specified in regulations under the Safe Drinking Water Act as the maximum permissible.

Driver Fuel — These fuel tubes or assemblies usually contain enriched uranium, plutonium, or thorium materials, which can be fissioned (or split) by neutrons. Because this fuel drives neutron bombardment of targets or blanket in a production, breeder, or research reactor, it is called driver fuel.

Dry Storage — Storage of spent nuclear fuel in environments where the fuel is not immersed in liquid for purposes of cooling and/or shielding.

Effective Dose Equivalent — The sum of the products of the dose equivalent received by specified tissues of the body and a tissue-specific weighting factor. This sum is a risk-equivalent value and can be used to estimate the health effects risk to the exposed individual. The tissue-specific weighting factor represents the fraction of the total health risk resulting from uniform whole-body irradiation that would be contributed by that particular tissue. The effective dose equivalent includes the committed effective dose equivalent from internal deposition of radionuclides, and the effective dose equivalent due to penetrating radiation from sources external to the body. Effective dose equivalent is expressed in units of rem or Sievert.

Effluent — Gas or fluid discharged into the environment.

Effluent (liquid) — Wastewater, treated or untreated, that flows out of a treatment plant, sewer, or industrial outfall; generally refers to waste discharged into surface waters.

Electrometallurgical Treatment — A technique to collect, concentrate, and immobilize fission products and transuranic elements from metallic spent nuclear fuel by removing the uranium in the spent fuel with an electrochemical cell. The treatment alters the chemical and physical nature of spent nuclear fuel to reduce its toxicity, volume, and mobility to render it amendable to transport, storage, or disposal.

Emergency Condition — For a nuclear facility, occurrences or accidents that might occur infrequently during startup testing or operation of the facility. Equipment, components, and structures might be deformed by these conditions to the extent that repair is required prior to reuse.

Emission — A material discharged into the atmosphere from a source operation or activity.

Emission Standards — Legally enforceable limits on the quantities and/or kinds of air contaminants that may be emitted into the atmosphere.

Empirical — Something that is based on actual measurement, observation, or experience rather than on theory.

Endangered Species — Any species which is in danger of extinction throughout all or significant portions of its range. The Endangered Species Act of 1973, as amended, establishes procedures for placing species on the Federal lists of endangered or threatened species.

Enriched Uranium — Uranium in which the abundance of the isotope uranium-235 is increased above the normal (naturally occurring) level of 0.711 weight percent.

Entrainment — The involuntary capture and inclusion of organisms in streams of flowing water; a term often applied to the cooling water systems of power plants/reactors. The organisms involved may include phyto- and zooplankton, fish eggs and larvae (ichthyoplankton), shellfish larvae, and other forms of aquatic life.

Environment, Safety, and Health Program — In the context of the U.S. Department of Energy (DOE), encompasses those DOE requirements, activities, and functions in the conduct of all DOE and DOE-controlled operations that are concerned with: impacts to the biosphere; compliance with environmental laws, regulations, and standards controlling air, water, and soil pollution; limiting the risks to the well-being of both the operating personnel and the general public; and protecting property against accidental loss or damage. Typical activities and functions related to this program include, but are not limited to, environmental protection, occupational safety, fire protection, industrial hygiene, health physics,

occupational medicine, process and facilities safety, nuclear safety, emergency preparedness, quality assurance, and radioactive and hazardous waste management.

Environmental Assessment — A written environmental analysis prepared pursuant to the National Environmental Policy Act. This assessment is performed to determine whether a Federal action could significantly affect the environment and thus require preparation of a more detailed environmental impact statement. If the action will not significantly affect the environment, then a Finding of No Significant Impact is prepared.

Environmental Impact Statement (EIS) — A document required of Federal agencies by the National Environmental Policy Act for major proposals or legislation significantly affecting the environment. A tool for decision making, it describes the positive and negative effects of the undertaking and alternative actions.

Environmental Justice — The fair treatment of people of all races, cultures, incomes, and educational levels with respect to the development, implementation, and enforcement of environmental laws, regulations, and policies. Fair treatment implies that no population of people should be forced to shoulder a disproportionate share of the negative environmental impacts of pollution or environmental hazards due to a lack of political or economic influence.

Environmental Survey — A documented, multi disciplined assessment (with sampling and analysis) of a facility to determine environmental conditions and to identify environmental problems requiring corrective action.

Epidemiology — The science concerned with the study of events that determine and influence the frequency and distribution of disease, injury, and other health-related events and their causes in a defined human population.

Equivalent Sound (Pressure) Level — The equivalent steady sound level that, if continuous during a specified time period, would contain the same total energy as the actual time varying sound. For example, L_{eq} (1-h) and L_{eq} (24-h) are the 1-hour and 24-hour equivalent sound levels, respectively.

Existing Facilities — Facilities that are projected to exist as of the Record of Decision for this environmental impact statement, scheduled for July 2000.

Exposure Limit — The level of exposure to a hazardous chemical (set by law or a standard) at which or below which adverse human health effects are not expected to occur:

(1) Reference dose is the chronic exposure dose (milligrams or kilograms per day) for a given hazardous chemical at which or below which adverse human noncancer health effects are not expected to occur.

(2) Reference concentration is the chronic exposure concentration (milligrams per cubic meter) for a given hazardous chemical at which or below which adverse human noncancer health effects are not expected to occur.

External Accident — Accidents initiated by manmade energy sources not associated with operation of a given facility. Examples include airplane crashes, induced fires, and transportation accidents adjacent to a facility.

Fault — A fracture or a zone of fractures within a rock formation along which vertical, horizontal, or transverse slippage has occurred. A normal fault occurs when the hanging wall has been depressed in

relation to the footwall. A reverse fault occurs when the hanging wall has been raised in relation to the footwall.

Finding of No Significant Impact — A document by a Federal agency briefly presenting the reasons why an action, not otherwise excluded, would not have a significant effect on the human environment and would not require an environmental impact statement under the National Environmental Policy Act.

Fissile Materials — Although sometimes used as a synonym for fissionable material, this term has acquired a more restricted meaning, namely, any material fissionable by thermal (slow) neutrons. The three primary fissile materials are uranium-233, uranium-235, and plutonium-239.

Fission (Fissioning) — The splitting of a nucleus into at least two other nuclei and the release of a relatively large amount of energy. Two or three neutrons are usually released during this type of transformation.

Fission Products — Elements formed by the fission of heavy elements (primary fission products); also, the elements formed by the decay of the primary fission products, many of which are radioactive.

Fissionable Material — Material that could undergo fission by the absorption of fast neutrons.

Fissium — An alloy of molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium that is designed to simulate fission products.

Floodplain — The lowlands adjoining inland and coastal waters and relatively flat areas.

Formation — In geology, the primary unit of formal stratigraphic mapping or description. Most formations possess certain distinctive features.

Fossil — Impression of trace of an animal or plant of past geological ages that has been preserved in the earth's crust.

Fuel Assembly — A cluster of fuel elements (or rods).

Fuel Element — Nuclear reactor component that includes the fissile material (fuel pin) sealed in cladding.

Fuel Pin — The uranium metal or alloy that undergoes fission in a nuclear reactor.

Fugitive Emissions — Emissions to the atmosphere from pumps, valves, flanges, seals, and other process points not vented through a stack. Also includes emissions from area sources such as ponds, lagoons, landfills, piles of stored material, and exposed soil.

g — A designator for ground motion acceleration, the rate of displacement of the ground due to the passage of elastic waves arising from earthquakes, explosions, seismic shots, machinery, wind, traffic, and other causes. The unit of acceleration is equal to about 9.8 meters (32.2 feet) per second^2.

Gamma-emitter — A radioactive substance that decays by releasing gamma radiation.

Gamma Rays — High-energy, short-wavelength, electromagnetic radiation accompanying fission and either emitted from the nucleus of an atom or emitted by some radionuclide or fission product. Gamma rays are very penetrating and can be stopped only by dense materials (such as lead) or a thick layer of shielding materials.

Gaussian Plume — The distribution of material (a plume) in the atmosphere resulting from the release of pollutants from a stack or other source. The distribution of concentrations about the centerline of the plume, which is assumed to decrease as a function of its distance from the source and centerline (Gaussian distribution), depends on the mean wind speed and atmospheric stability.

Genetic Effects — The outcome resulting from exposure to mutagenic chemicals or radiation which results in genetic changes in germ line or somatic cells. There are two kinds of genetic effects:

(1) Effects on genetic material in reproductive cells that cause trait modifications that can be passed from parents to offspring.

(2) Effects on genetic material in nonreproductive cells that result in tissue or organ modifications (e.g., liver tumors) that do not pass from parents to offspring.

Geologic Repository — A system that is intended to be used for, or may be used for, the disposal of radioactive waste or spent nuclear fuel in excavated geologic media. A geologic repository includes (1) the geologic repository operations area, and (2) the portion of the geologic setting that provides isolation. A near-surface disposal area is not a geologic repository.

Geology — The science that deals with the Earth: the materials, processes, environments, and history of the planet, including the rocks and their formation and structure.

Groundwater — The supply of water found beneath the Earth's surface, usually in aquifers, which may supply wells and springs.

Habitat — The environment occupied by individuals of a particular species, population, or community.

Half-Life — The time in which half the atoms of a radioactive isotope decay to another nuclear form. Half-lives vary from millionths of a second to billions of years.

Hazardous Chemical — Under 29 CFR 1910, Subpart Z, "hazardous chemicals" are defined as "any chemical which is a physical hazard or a health hazard." Physical hazards include combustible liquids, compressed gases, explosives, flammables, organic peroxides, oxidizers, pyrophorics, and reactives. A health hazard is any chemical for which there is good evidence that acute or chronic health effects occur in exposed employees. Hazardous chemicals include carcinogens, toxic or highly toxic agents, reproductive toxins, irritants, corrosives, sensitizers, hepatotoxins, nephrotoxins, agents that act on the hematopoietic system, and agents that damage the lungs, skin, eyes, or mucous membranes.

Hazardous Material — A material, including a hazardous substance, as defined by 49 CFR 171.8, which poses a risk to health, safety, and property when transported or handled.

Hazardous Substance — Any substance that when released to the environment in an uncontrolled or unpermitted fashion becomes subject to the reporting and possible response provisions of the Clean Water Act and the Comprehensive Environmental Response, Compensation, and Liability Act.

Hazardous/Toxic Air Pollutants — Air pollutants known or suspected to cause serious health problems such as cancer, poisoning, or sickness, and may have immunological, neurological, reproductive, developmental, or respiratory effects.

Hazardous Waste — Any solid waste (can also be semisolid or liquid, or contain gaseous material) having the characteristics of ignitability, corrosivity, toxicity, or reactivity, defined by the Resource Conservation and Recovery Act and identified or listed in 40 CFR 261 or by the Toxic Substances Control Act.

Heavy Metals — Metallic or semimetallic elements of high molecular weight, such as mercury, chromium, cadmium, lead, and arsenic, that are toxic to plants and animals at known concentrations.

High Efficiency Particulate Air Filter — A filter used to remove very small particulates from dry gaseous effluent streams.

High-Level Radioactive Waste — The (1) highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid waste derived from such liquid waste that contains fission products in sufficient concentrations; and (2) other highly radioactive material that is determined, consistent with existing law, to require permanent isolation.

Historic Resources — Archaeological sites, architectural structures, and objects produced after the advent of written history dating to the time of the first Euro-American contact in an area.

Hot Cell/Hot Cell Facility — A heavily shielded enclosure for handling and processing (by remote means or automatically), or storing highly radioactive materials.

Impingement — The process by which aquatic organisms too large to pass through the screens of a water intake structure become caught on the screens and are unable to escape.

Inert cell — An enclosure where operations that require very low oxygen levels are performed.

Ingot — A mass of metal cast in a standard shape for convenient storage or shipment.

Involved Worker — Workers that would be involved in a proposed action as opposed to workers that would be on the site of a proposed action but not involved in the action.

Ionizing Radiation — Alpha particles, beta particles, gamma rays, neutrons, high-speed electrons, high-speed protons, and other particles or electromagnetic radiation that can displace electrons from atoms or molecules, thereby producing ions.

Isotope — An atom of a chemical element with a specific atomic number and atomic mass. Isotopes of the same element have the same number of protons, but different numbers of neutrons and different atomic masses.

Joule — A metric unit of energy, work, or heat, equivalent to 1 watt-second, 0.737 foot-pound, or 0.239 calories.

Karst Terrain — A type of land surface that is found in regions underlain by soluble rocks, such as limestone and dolomite, which is peculiar to dependent upon underground solution of the bedrock and the diversion of the surface waters to underground waters (that is, stream that disappear underground). Karst terrain is characterized by sinkholes, underground streams, and caves.

Landscape Character — The arrangement of a particular landscape as formed by the variety and intensity of the landscape features (land, water, vegetation, and structures) and the four basic elements (form, line,

color, and texture). These factors give an area a distinctive quality that distinguishes it from its immediate surroundings.

Latent Cancer Fatalities — Fatalities associated with acute or chronic environmental exposure to chemicals or radiation that occur from delayed effects years after exposure.

Liquid Metal Cooled Breeder Reactor — A reactor that creates more fissionable material than it consumes and uses liquid metal as a coolant. Liquid sodium is a common metal used to cool this type of reactor.

Long-Lived Isotopes — Radionuclides with half-lives greater than about 30 years.

Long-term Storage — The storage of hazardous waste (a) onsite (a generator site) for a period of 90-days or greater, other than in a satellite accumulation area, or (b) offsite in a properly managed treatment, storage, or disposal facility for any period of time.

Low-Level Radioactive Waste — Waste that contains radioactivity, but is not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or by-product material as defined by Section 11e (2) of the Atomic Energy Act of 1954, as amended.

Management — As used in this EIS, the stabilization and interim storage of sodium-bonded spent nuclear fuel pending final disposition.

Maximum Contaminant Level — The maximum permissible level of a contaminant in water delivered to any user of a public drinking water system. Maximum contaminant levels are enforceable standards under the Safe Drinking Water Act.

Maximally Exposed Individual (MEI) — A hypothetical individual defined to allow dose or dosage comparison with numerical criteria for the public. This individual is located at the point on the U.S. Department of Energy site boundary nearest to the facility in question. A hypothetical person who could potentially receive the maximum dose of radiation or hazardous chemicals.

Megajoule — A unit of heat, work, or energy equal to 1 million joules. See "Joule."

Meteorology — The science dealing with the atmosphere and its phenomena, especially as relating to weather.

Metric Tons of Heavy Metal — Quantities of unirradiated and spent nuclear fuel 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 movement of a material through the air, soil, or groundwater; also, seasonal movement of animals from one area to another.

Millirem — One thousandth of a rem.

Mixed Waste — Waste that contains both a hazardous waste subject to Resource Conservation and Recovery Act, and source, special nuclear or by-product material subject to the Atomic Energy Act of 1954 (42 U.S.C. 2011 *et seq.*).

Mollusks — Unsegmented, invertebrate animals including gastropods, pelecypods, and cephalopods.

National Ambient Air Quality Standards (NAAQS) — Uniform, national air quality standards established by the Environmental Protection Agency under the authority of the Clean Air Act that restrict ambient levels of criteria pollutants to protect public health (primary standards) or public welfare (secondary standards), including plant and animal life, visibility, and materials. Standards have been set for ozone, carbon monoxide, particulates, sulfur dioxide, nitrogen, nitrogen dioxide, and lead.

National Emission Standards for Hazardous Air Pollutants — A set of national emission standards for listed hazardous pollutants emitted from specific classes or categories of new and existing sources.

National Pollutant Discharge Elimination System (NPDES) — Federal permitting system required for water pollution effluent under the Clean Water Act, as amended.

National Register of Historic Places — A list maintained by the Secretary of the Interior of districts, sites, buildings, structures, and objects of prehistoric or historic local, state, or national significance under Section 2(b) of the Historic Sites Act of 1935 (16 U.S.C. 462) and Section 101(a) (1) (A) of the National Historic Preservation Act of 1966, as amended.

Neutron — An uncharged elementary particle with a mass slightly greater than that of the proton, found in the nucleus of every atom heavier than hydrogen-1. A free neutron is unstable and decays with a half-life of about 13 minutes into an electron and a proton; used in the fission process.

Neutron Flux — The product of neutron number density and velocity (energy), giving an apparent number of neutrons flowing through a unit area per unit time.

Neutron Poison — A material (e.g., a boron or a burnable absorber rod) inserted into a nuclear reactor or spent fuel pool to absorb neutrons and end criticality. Any material with a strong affinity for absorbing neutrons without generating new neutrons that can be used to control the nuclear chain reaction.

Nitrogen Oxides — Refers to the oxides of nitrogen, primarily nitrogen oxide (NO) and nitrogen dioxide (NO_2). These are produced in the combustion of fossil fuel and can constitute an air pollution problem. Nitrogen dioxide emissions contribute to acid deposition and formation of atmospheric ozone.

Noble metals — A group of metals such as zirconium, niobium, and gold that are highly resistant to oxidation and corrosion.

Noise — Any sound that is undesirable because it interferes with speech and hearing, or is intense enough to damage hearing, or is otherwise annoying (unwanted sound).

Nonattainment Area — An air quality control region (or portion thereof) in which the Environmental Protection Agency has determined that ambient air concentrations exceed national ambient air quality standards for one or more criteria pollutants.

Normal Conditions — All activities associated with a facility mission, whether operation, maintenance, storage, and so forth, which are carried out within a defined envelope. This envelope can be design process conditions, performance in accordance with procedures, and so forth.

Notice of Intent — Announces the scoping process. The Notice of Intent is usually published in the Federal Register and a local newspaper. The scoping process includes holding at least one public meeting and requesting written comments on what issues and environmental concerns an environmental impact statement should address.

Nuclear Power Plant — A facility that converts nuclear energy into electrical power.

Nuclear Radiation — Particles (alpha, beta, neutrons) or photons (gamma) emitted from the nucleus of unstable radioactive atoms as a result of radioactive decay.

Nuclear Reaction — A reaction in which an atomic nucleus is transformed into another isotope of that respective nuclide, or into another element altogether; it is always accompanied by the liberation of either particles or energy.

Nuclear Reactor — A device that sustains a controlled nuclear fission chain reaction that releases energy in the form of heat.

Nuclear Regulatory Commission (NRC) — The Federal agency that regulates the civilian nuclear power industry in the United States.

Nuclide — A species of atom characterized by the constitution of its nucleus and, hence, by the number of protons, the number of neutrons, and the energy content.

Obsidian — Volcanic glass, which may be banded and is usually black in color, although examples in red, green, and brown are also known.

Occupational Safety and Health Administration (OSHA) — A Federal agency that oversees and regulates workplace health and safety; created by the Occupational Safety and Health Act of 1970.

Off-gas — A volatile and semivolatile gaseous product that is released during a process.

Off Site — As used in the environmental impact statement, the term denotes a location, facility, or activity occurring outside of the boundary of the facility of interest.

Ozone — The triatomic form of oxygen; in the stratosphere, ozone protects the Earth from the sun's ultraviolet rays, but in lower levels of the atmosphere, ozone is considered an air pollutant.

Packaging — With regard to hazardous or radionuclide materials, the assembly of components necessary to ensure compliance with Federal regulations for transportation. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle tie-down system and auxiliary equipment may be designated as part of the packaging.

Particulate Matter — Air pollutants including dust, dirt, soot, smoke, or liquid droplets emitted into the air. "Total suspended particulate" was first used as the indicator for particulate concentrations. Current standards use the indicators "PM_{10}" and "$PM_{2.5}$," which include only those particles with an aerodynamic diameter smaller than or equal to 10 micrometers and 2.5 micrometers, respectively. The smaller particles are more responsible for adverse health effects because they reach further into the respiratory tract.

Permutation — Changing the order of elements arranged in a particular order.

Person-Rem — The unit of collective radiation dose to a given population; the sum of the individual doses received by a population segment.

Playa — A dry lake bed in a desert basin or a closed depression that contains water on a seasonal basis.

Plume — A flowing, often somewhat conical, trail of emissions from a continuous point source.

Plume Immersion — With regard to radiation, the situation in which an individual is enveloped by a cloud of radiation gaseous effluent and receives an external radiation dose.

Plutonium — A heavy, radioactive, metallic element with the atomic number 94. It is produced artificially in a reactor by bombardment of uranium with neutrons and is used in the production of nuclear weapons.

Poison — See "neutron poison."

Pounds per Square Inch — A measure of pressure; atmospheric pressure is about 14.7 pounds per square inch.

Prevention of Significant Deterioration — An Environmental Protection Agency program, mandated by the Clean Air Act, in which state or Federal permits are required that are intended to limit increases in air pollutant concentrations by restricting emissions for new or modified sources in places where air quality is already better than required to meet primary and secondary ambient air quality standards.

Prime Farmland — Land that has the best combination of physical and chemical characteristics for producing food, feed, fiber, forage, oil-seed, and other agricultural crops with minimum inputs of fuel, fertilizer, pesticides, and labor without intolerable soil erosion, as determined by the Secretary of Agriculture (Farmland Protection Act of 1981, 7 CFR 7, paragraph 658).

Probabilistic Risk Assessment — A comprehensive, logical, and structured methodology to identify and quantitatively evaluate significant accident sequences and their consequences.

Probable Maximum Flood — Flood levels predicted for a scenario having hydrological conditions that maximize the flow of surface waters.

Programmatic Environmental Impact Statement — A legal document prepared in accordance with the requirements of Section 102(2)(C) of the National Environmental Policy Act which evaluates the environmental impacts of proposed Federal actions that involve multiple decisions potentially affecting the environment at one or more sites.

Proliferation (Nuclear) — The spread of nuclear weapons and the materials and technologies used to produce them.

PUREX (Plutonium Uranium Extraction) — A chemical separation process that has been used for recovering uranium and plutonium from irradiated fuel in a form usable as reactor fuel or for weapons. The process uses aqueous solvent extraction to perform the separation. This technology can also be used to treat spent nuclear fuel for disposal.

Pyrophoric — Being highly susceptible to spontaneous ignition and continuous combustion.

Qualitative Environmental Impacts — 10 CFR 51, Appendix B defines the qualitative terms "small," "moderate," and "large" as follows:

Small	Environmental effects are not detectable or are so minor that they would neither destabilize nor noticeably alter any important attribute of the resource. For the purposes of assessing radiological impacts, the U.S. Nuclear Regulatory

Commission (NRC) has concluded that those impacts that do not exceed permissible levels in the NRC's regulations are considered small.

Moderate — Environmental effects are sufficient to alter noticeably, but not to destabilize, important attributes of the resource.

Large — Environmental effects are clearly noticeable and are sufficient to destabilize important attributes of the resource.

Quality Factor — The principal modifying factor that is employed to derive dose equivalent from absorbed dose.

Rad — See "radiation absorbed dose."

Radiation — The emitted particles or photons from the nuclei of radioactive atoms. Some elements are naturally radioactive; others are induced to become radioactive by bombardment in a reactor.

Radiation Absorbed Dose (rad) — The basic unit of absorbed dose equal to the absorption of 0.01 Joule per kilogram of absorbing material.

Radioactive Mixed Waste — Waste containing both radioactive and hazardous components regulated by the Atomic Energy Act and the Resource Conservation and Recovery Act, respectively. The term "radioactive component: refers only to the actual radionuclides dispersed or suspended in the waste substance.

Radioactive Waste — Materials from nuclear operations that are radioactive or are contaminated with radioactive materials, and for which use, reuse, or recovery are impractical.

Radioactivity — The spontaneous decay or disintegration of unstable atomic nuclei, accompanied by the emission of radiation.

Radioisotopes — Radioactive nuclides of the same element (same number of protons in their nuclei) that differ in the number of neutrons.

Radionuclide — A radioactive element characterized according to its atomic mass and atomic number which can be man-made or naturally occurring.

Radon — Gaseous, radioactive element with the atomic number 86 resulting from the radioactive decay or radium. Radon occurs naturally in the environment, and can collect in unventilated enclosed areas, such as basements. Large concentrations of radon can cause lung cancer in humans.

RADTRAN — A computer code that combines user-determined meteorological, demographic, transportation, packaging, and material factors with health physics data to calculate the expected radiological consequences and accident risk of transporting radioactive material.

Reactive — Having low chemical stability and subject to high chemical reaction rates.

Record of Decision — A document prepared in accordance with the requirements of the Council on Environmental Quality and National Environmental Policy Act regulations 40 CFR 1505.2, that provides a concise public record of the decision on a proposed Federal action for which an environmental impact statement was prepared. A Record of Decision identifies the alternatives considered in reaching the decision,

the environmentally preferable alternative(s), factors balanced in making the decision, whether all practicable means to avoid or minimize environmental harm have been adopted, and if not, why they were not.

Regional Economic Area — A geographic area consisting of an economic node and the surrounding counties that are economically related and include the places of work and residences of the labor force. Each regional economic area is defined by the U.S. Bureau of Economic Analysis.

Region of Influence — A site-specific geographic area that includes the counties where approximately 90 percent of the current U.S. Department of Energy and/or contractor employees reside.

Rem — See "roentgen equivalent man."

Remediation — The process, or a phase in the process, of rendering radioactive, hazardous, or mixed waste environmentally safe, whether through processing, entombment, or other methods.

Reprocessing (of spent nuclear fuel) — Processing of reactor-irradiated nuclear material (primarily spent nuclear fuel) to recover fissile and fertile material, to recycle such materials primarily for defense programs. Historically, reprocessing has involved aqueous chemical separation of elements (typically uranium or plutonium) from undesired elements in the fuel.

Riparian — Of, on, or relating to the banks of a natural course of water.

Risk — A quantitative or qualitative expression of possible loss that considers both the probability that a hazard will cause harm and the consequences of that event.

Risk Assessment (chemical or radiological) — The qualitative and quantitative evaluation performed in an effort to define the risk posed to human health and/or the environment by the presence or potential presence and/or use of specific chemical or radiological materials.

Roentgen — A unit of exposure to ionizing X or gamma radiation equal to or producing 1 electrostatic unit of charge per cubic centimeter of air. It is approximately equal to 1 rad.

Roentgen Equivalent Man (rem) — A measure of radiation dose (i.e., the average background radiation dose is 0.3 rem per year). The unit of biological dose equal to the product of the absorbed dose in rads; a quality factor, which accounts for the variation in biological effectiveness of different types of radiation; and other modifying factors.

Runoff — The portion of rainfall, melted snow, or irrigation water that flows across the ground surface and eventually enters streams.

Safety Analysis Report — A document that provides a complete description and safety analysis of a facility design, normal and emergency operations, hypothetical accidents and their predicted consequences, and the means proposed to prevent such accidents or mitigate their consequences.

Safety Evaluation Report — A document prepared by the U.S. Nuclear Regulatory Commission that evaluates documentation (i.e., technical specifications, safety analysis reports, and special safety reviews and studies) submitted by a licensee for its approval. This ensures that all of the safety aspects of part or all of the activities conducted at the facility are formally and thoroughly analyzed, evaluated, and recorded.

Sanitary waste — Waste generated by normal housekeeping activities, liquid or solid (including sludge), which are not hazardous or radioactive.

Scope — In a document prepared pursuant to the National Environmental Policy Act of 1969, the range of actions, alternatives, and impacts to be considered.

Scoping — The solicitation of comments from interested persons, groups, and agencies at public meetings, public workshops, in writing, electronically, or via fax to assist in defining the proposed action, identifying alternatives, and developing preliminary issues to be addressed in an environmental impact statement.

Seismic — Pertaining to any Earth vibration, especially an earthquake.

Seismic Zone — An area defined by the Uniform Building Code (1991), designating the amount of damage to be expected as the result of earthquakes. The United States is divided into six zones: (1) Zone 0: no damage; (2) Zone 1: minor damage, corresponds to intensities V and VI of the modified Mercalli intensity scale; (3) Zone 2A: moderate damage, corresponds to intensity VII of the modified Mercalli intensity scale (eastern U.S.); (4) Zone 2B: slightly more damage than 2A (western U.S.); (5) Zone 3: major damage, corresponds to intensity VII and higher of the modified Mercalli intensity scale; (6) Zone 4: areas within Zone 3 determined by proximity to certain major fault systems.

Severe Accident — An accident with a frequency rate of less than 10^{-6} per year that would have more severe consequences than a design-basis accident, in terms of damage to the facility, offsite consequences, or both.

Sewage — The total of organic waste and wastewater generated by an industrial establishment or a community.

Shielding — With regard to radiation, any material of obstruction (bulkheads, walls, or other construction) that absorbs radiation in order to protect personnel or equipment.

Short-Lived Nuclides — Radioactive isotopes with half-lives no greater than about 30 years (e.g., cesium-137 and strontium-90).

Shutdown — For a U.S. Department of Energy (DOE) reactor, that condition in which the reactor has ceased operation and DOE has declared officially that it does not intend to operate it further (see DOE Order 5480.6, *Safety of Department of Energy-Owned Nuclear Reactors*).

Silt — A sedimentary material consisting of fine mineral particles intermediate in size between sand and clay.

Sinkhole — A depression in the earth's surface formed by the collapse of a cavern roof. Typically associated with Karst terrain.

Sodium-bonded — Physically in contact with and attached to the element sodium.

Source Term — The estimated quantities of radionuclides or chemical pollutants available for release to the environment.

Species of Special Concern — Native species that are either low in numbers, limited in distribution, or have suffered significant habitat losses.

Spent Nuclear Fuel — Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated for reprocessing.

Standardized Canister —As used in this EIS, this refers to a standard-sized U.S. Department of Energy canister which is stainless steel, right circular cylinder with a nominal outside diameter of 45.7 centimeters (18 inches), or 61 centimeters (24 inches), a nominal thickness of 0.95 centimeters (0.375 inches), or 1.27 centimeters (0.50 inches), respectively. Each canister could have a maximum overall length of either 3 meters (118.11 inches) with a useable length of 2.25 meters (88.58 inches), or 4.57 meters (179.92 inches) with useable length of 4.11 meters (161.81 inches). The standardized canister is used for storing spent nuclear fuel assemblies, high-integrity cans, and any other waste packages.

Surface Water — Water on the Earth's surface, as distinguished from water in the ground (groundwater).

Threatened Species — Any species designated under the Endangered Species Act as likely to become an endangered species within the foreseeable future throughout all or a significant portion of its range.

Threshold Limit Values — The recommended highest concentrations of contaminants to which workers may be exposed according to the American Conference of Governmental Industrial Hygienists.

Transuranic Waste — Waste contaminated with alpha-emitting radionuclides with half-lives greater than 20 years and concentrations greater than 100 nanocuries/gram at time of assay. A nanocurie is 10^{-9} curies.

Treatment — In this EIS, a process to remove and/or stabilize metallic sodium.

Unusual Occurrence — Any unusual or unplanned event that adversely affects or potentially affects the performance, reliability, or safety of a facility.

Uranium — A heavy, silvery-white metallic element (atomic number 92) with several radioactive isotopes that is used as fuel in nuclear reactors or as radiation shielding.

Viewshed — The extent of an area that may be viewed from a particular location. Viewsheds are generally bounded by topographic features such as hills or mountains.

Visual Resource Management Class — A class defines the different degrees of modification allowed to the basic elements of landscape. They are: Class I – Applied to wilderness areas, wild and scenic rivers, and other similar situations - the level of change to the characteristic landscape should be very low and must not attract attention; Class II – Management activities may be seen, but do not attract attention; Class III – Management activities may attract attention, but should not dominate the view; Class IV – Management activities may dominate the view.

Vitreous — Resembling or having the nature of glass.

Vitrification — The process of immobilizing waste material that results in glass-like solid.

Volatile Organic Compounds — A broad range of organic compounds that vaporize at ambient or relatively low temperatures, such as benzene, chloroform, and methyl alcohol. With regard to air pollution, any organic compound that participates in atmospheric photochemical reaction, except for those designated by the Environmental Protection Agency administrator as having negligible photochemical reactivity.

Waste Minimization and Pollution Prevention — An action that economically avoids or reduces the generation of waste and pollution by source reduction, reducing the toxicity of hazardous waste and pollution, improving energy use, or recycling. These actions will be consistent with the general goal of minimizing present and future threats to human health, safety, and the environment.

Weighting Factor — With regard to radiation, the fraction of the total health risk resulting from uniform whole-body irradiation that could be contributed to that particular tissue.

Wetlands — Those areas that are inundated or saturated by surface or ground water at a frequency and duration sufficient to support, and that under normal circumstances do support, a prevalence of vegetation typically adapted for life in saturated soil conditions. Wetlands generally include swamps, marshes, bogs, and similar areas.

Whole-Body Dose — With regard to radiation, the dose resulting from the uniform exposure of all organs and tissues in a human body. (Also see "effective dose equivalent.")

Wind Rose — A depiction of wind speed and direction frequency for a given period of time.

X/Q (Chi/Q) — The relative calculated air concentration due to a specific air release and atmospheric dispersion; units are (seconds per cubic meter). For example (curies per cubic meter)/(curies per second) = (seconds per cubic meter) or (grams per cubic meter)/(grams per second) = (seconds per cubic meter).

Zeolite — One of approximately 30 hydrous (water containing) aluminum silicate minerals or their corresponding synthetic compounds, used chiefly as molecular filters and ion-exchange agents such as is used in a water softener. It is used in electrometallurgical treatment to collect and contain fission products from process salt.

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8. DISTRIBUTION LIST

The U.S. Department of Energy is providing copies of the final environmental impact statement (EIS) (or Summary) to Federal, state, and local elected and appointed officials and agencies of government; Native American representatives; 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.

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Ralph M. Hall (Committee on Science)
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Federal Agencies

Advisory Council on Historic Preservation
Defense Nuclear Facilities Safety Board
Department of Agriculture
Department of Defense
Department of Interior
Department of State
Department of Transportation
Environmental Protection Agency
National Science Foundation
Nuclear Regulatory Commission
Office of Science and Technology Policy

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Approximately 300 copies of the Final EIS were sent to stakeholders
Approximately 1,500 copies of the Summary of the Final EIS were sent to stakeholders

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