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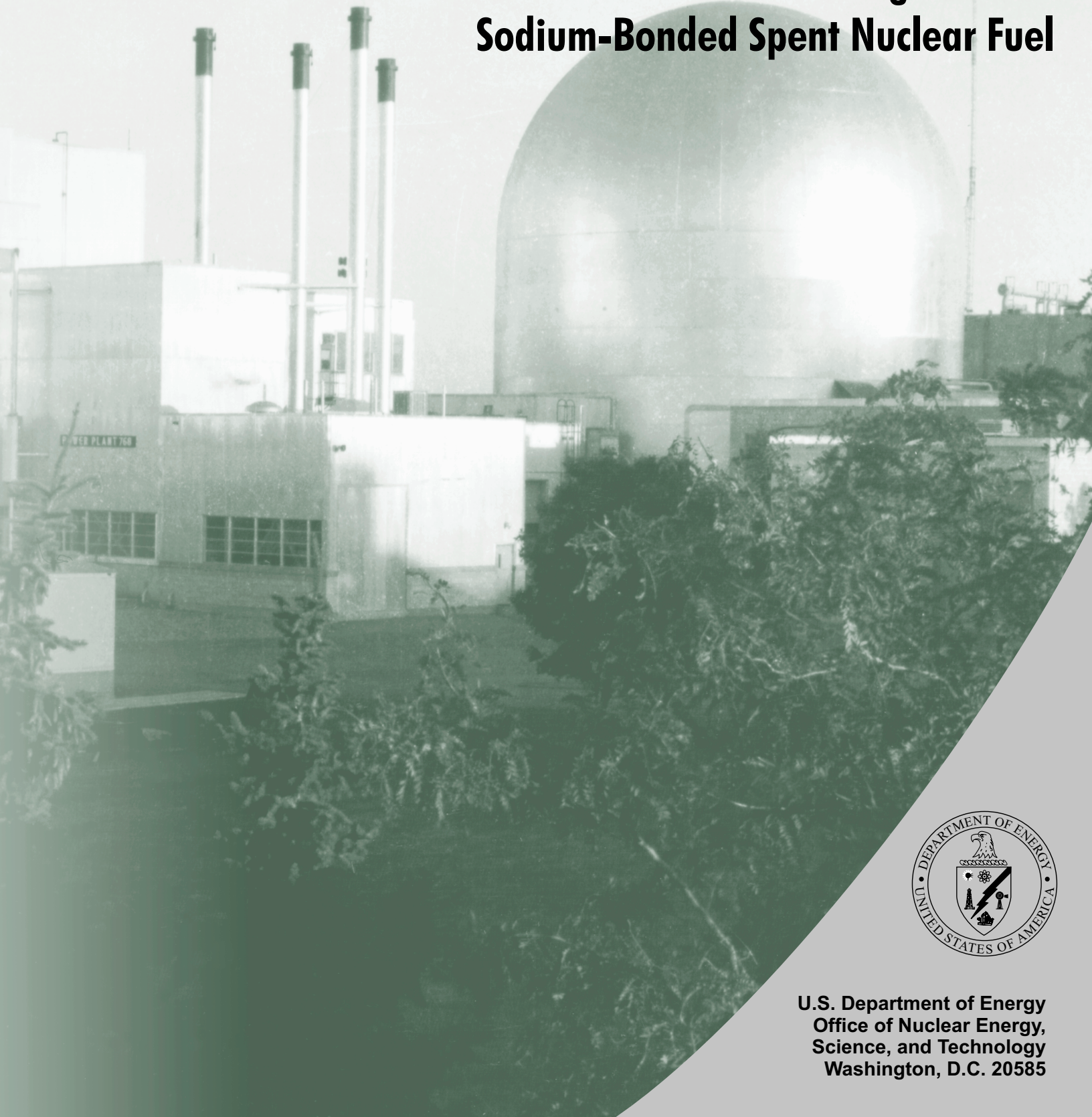
Summary

Draft Environmental Impact Statement

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



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Abstract: The Department of Energy (DOE) is responsible for the safe and efficient management of several different types of spent nuclear fuel. One type of spent nuclear fuel that may not be suitable for disposal in a geologic repository without treatment is the DOE-owned sodium-bonded spent nuclear fuel. Sodium-bonded spent nuclear 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 material could complicate the process of qualifying and licensing such spent nuclear fuel for disposal in a geologic repository. Currently, more than 98 percent of DOE's sodium-bonded spent nuclear fuel is located at the Idaho National Engineering and Environmental Laboratory (INEEL). In a 1995 agreement with the State of Idaho, DOE committed to remove all spent nuclear fuel from Idaho by 2035.

Several technologies for spent nuclear fuel treatment are under development and might facilitate qualification and licensing for ultimate disposal. The most developed technology is the electrometallurgical treatment of sodium-bonded spent nuclear fuel at Argonne National Laboratory-West (ANL-W). This EIS evaluates the potential environmental impacts associated with the treatment of sodium-bonded spent nuclear fuel in one or more spent nuclear fuel management facilities: ANL-W at INEEL (near Idaho Falls, Idaho) and either the F-Canyon or Building 105-L at the Savannah River Site (near Aiken, South Carolina). The EIS analyzes under the proposed action the electrometallurgical process, the plutonium-uranium extraction (PUREX) process, direct disposal in high-integrity cans with the sodium removed, and the melt and dilute process. The EIS also evaluates the continued storage of sodium-bonded spent nuclear fuel and direct disposal without treatment under the No Action Alternative.

Public Comments: In preparing this Draft EIS, DOE considered comments received from the public during the scoping process (February 22, 1999 to April 8, 1999). Comments on this Draft EIS may be submitted during the 45-day comment period. Public meetings on this EIS will also be held during the comment period. The dates, times, and locations of these meetings will be announced shortly after issuance of this Draft EIS.

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ACRONYMS AND ABBREVIATIONS

ANL	Argonne National Laboratory
ANL-W	Argonne National Laboratory-West
CFR	Code of Federal Regulations
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
DOE	U.S. Department of Energy
EBR-II	Experimental Breeder Reactor-II
EIS	Environmental Impact Statement
EMT	Electrometallurgical treatment (of spent fuel)
EPA	U.S. Environmental Protection Agency
FR	*Federal Register*
GMODS	Glass Material Oxidation and Dissolution System
IAEA	International Atomic Energy Agency
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
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
P.L.	Public Law
PUREX	Plutonium-Uranium Extraction
RCRA	Resource Conservation and Recovery Act
SRS	Savannah River Site
U.S.C.	United States Code

SUMMARY

This document summarizes the U.S. Department of Energy's *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*. In addition to information concerning the background, purpose and need for the proposed action, and the National Environmental Policy Act process, this summary describes the characteristics of sodium-bonded spent nuclear fuel, the proposed treatment methods, the proposed facilities, the alternatives considered, and the environmental consequences of these alternatives. A glossary is included at the end to assist the reader with some of the technical terms used in this document.

S.1 BACKGROUND

The U.S. Department of Energy's (DOE) *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (SBSNF EIS) identifies potential alternatives and impacts associated with the proposed treatment and management of DOE-owned sodium-bonded spent nuclear fuel and facilitate its disposal in a geologic repository. This environmental impact statement (EIS) was prepared in accordance with the National Environmental Policy Act (NEPA), as amended.

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) near Idaho Falls, Idaho; the Enrico Fermi Atomic Power Plant at 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 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, which is 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 to facilitate 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. These include: (1) an electrometallurgical treatment process; (2) the plutonium-uranium extraction (PUREX) process; (3) placement of the spent nuclear fuel in high-integrity cans; (4) a melt and dilute process; (5) a glass material oxidation and dissolution system (GMODS) process; (6) a direct plasma arc-vitreous ceramic process; and (7) 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[1], the final acceptance criteria will be more refined. If the repository is developed, final acceptance criteria will not be available until after DOE receives its construction authorization

[1] *Civilian Radioactive Waste Management System - Waste Acceptance System Requirements Document, 1998.*

from the U.S. Nuclear Regulatory Commission (NRC), based on the successful demonstration of the safe, long-term performance of the repository in accordance with the NRC regulations. Until such time, the preliminary acceptance criteria will tend 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 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 Statement* (Programmatic Spent Nuclear Fuel EIS), in which DOE decided to regionalize the management of DOE-owned spent nuclear fuel by fuel type. DOE also decided to: (1) continue environmental restoration activities at Idaho National Engineering and Environmental Laboratory (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. This Record of Decision was partially based on the conclusions of 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. It also analyzed the consequences of 10 years of waste and spent nuclear fuel management and environmental restoration actions at Idaho National Engineering Laboratory.[2]

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.), an d *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.

S.1.1 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 complicate the disposal qualification for the ultimate disposal of this spent nuclear fuel in a geologic repository. Metallic sodium reacts vigorously with water or moist air, 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 which used these fuels.

To the extent possible, sodium was removed from the external surfaces of these fuels after their 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 repository acceptance criteria probably will exclude reactive materials unless their packaging minimizes the probability of rapid oxidation. Finally, some of the sodium-bonded spent nuclear fuel contains highly enriched uranium, and its disposal in a geologic repository may require special criticality control measures.

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

The presence of reactive or pyrophoric materials such as metallic sodium and metallic uranium, or the presence of highly enriched uranium, could complicate the process of qualifying the spent nuclear fuel for disposal. Such qualification would require sufficient data and predictive analyses to demonstrate that emplacement of the spent nuclear fuel would not adversely affect a repository's ability to protect the environment and worker and public health and safety.

To ensure that the State of Idaho Settlement Agreement terms 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 qualifications. 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 fuels. 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.

It is prudent to evaluate these alternative treatment technologies now, while DOE is performing site characterization activities for a potential geologic repository. 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 to promote consistency between the two efforts and to minimize programmatic risks associated with waste form qualification and acceptance for ultimate disposal.

Geologic Repository and Waste Acceptance Criteria

Geologic repositories are deep, excavated underground vaults constructed for the purpose of permanently containing nuclear wastes. Any spent nuclear fuel packaging or treatment technology must be capable of putting fuel in a form that will satisfy acceptance criteria requirements. DOE has drafted preliminary acceptance criteria which are being used to assess the feasibility of DOE spent nuclear fuel disposition options. The draft criteria states that spent nuclear fuel containing materials that are explosive, pyrophoric, or chemically reactive in the repository environment would not meet the acceptance criteria. Because it contains metallic sodium, the sodium-bonded spent nuclear fuel could be categorized as hazardous waste that is potentially both pyrophoric and reactive.

These criteria would become more detailed, consistent with detailed designs of repository facilities and waste package performance, and finalized after the NRC has issued a construction authorization for a proposed repository. In order to ensure that the treatment option DOE could select will produce 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.

It is prudent to evaluate alternative treatment technologies now, while DOE is performing site characterization activities for a potential geologic repository at Yucca Mountain. 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 to promote consistency between the two efforts and to minimize programmatic risks associated with waste form qualification and acceptance for ultimate disposal.

S.1.2 National Environmental Policy Act (NEPA) Process

Prior to this EIS, an environmental assessment was prepared for the Electrometallurgical Treatment Research and Demonstration Project at Argonne National Laboratory-West (ANL-W). A Finding of No Significant Impact was issued in May 1996. The finding required preparation of an EIS if electrometallurgical treatment would be proposed to treat the remaining EBR-II spent nuclear fuel, or if electrometallurgical treatment or another technology would be proposed on a production scale for the remaining sodium-bonded spent nuclear fuel owned by DOE. DOE is currently evaluating its options for the treatment and management of sodium-bonded spent nuclear fuel. A key element of DOE's decision-making is a thorough understanding of the environmental impacts that may occur during the implementation of the proposed action. The National Environmental Policy Act of 1969, as amended, provides Federal agency decision-makers with a process to consider potential environmental consequences (both positive and negative) of proposed actions before agencies make decisions. In following this process, DOE has prepared this Draft EIS to assess various

alternatives and to provide the necessary background, data, and analyses to help decision-makers and the public understand the potential environmental impacts of each alternative.

National Environmental Policy Act

National Environmental Policy Act of 1969: A law that requires Federal agencies to consider in their decision-making processes the potential environmental effects of proposed actions and analyses of alternatives and measures to avoid or minimize the adverse effects of a proposed action.

Alternatives: A range of reasonable options considered in selecting an approach to meeting the proposed objectives. In accordance with other applicable requirements, the No Action Alternative is also considered.

Environmental Impact Statement: A detailed environmental analysis for a proposed major Federal action that could significantly affect the quality of the human environment. A tool to assist in decision-making, it describes the positive and negative environmental effects of the proposed undertaking and alternatives.

Record of Decision: A concise public record of DOE's decision, which discusses the decision, identifies the alternatives (specifying which ones were considered environmentally preferable), and indicates whether all practicable means to avoid or minimize environmental harm from the selected alternative were adopted (and if not, why not).

DOE's strategy for compliance with NEPA has been first to make decisions on programmatic alternatives in the Programmatic Spent Nuclear Fuel EIS, followed by site-specific analyses to implement the programmatic decisions.

Before an EIS can be prepared, the scope, i.e., the range of actions, alternatives, and impacts to be considered, must be determined. The NEPA process requires public participation in determining the scope of an EIS. The scoping process is initiated by a Federal agency's publication of a Notice of Intent to prepare an EIS in the *Federal Register*. DOE NEPA regulations require at least one public meeting and a minimum 30-day comment period to receive public input on the scope of the EIS.

S.1.2.1 Issues Identified During 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 issues to be resolved in 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.

Approximately 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 during the public scoping period are addressed below.

Many commentors at the public 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 Independent Assessment Final Report on the demonstration project; a nonproliferation assessment report; and an independent cost study. Several commentors said that this EIS is premature because the electrometallurgical treatment 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 independent nonproliferation assessment report in the same time frame as the Draft EIS, or that this EIS should be delayed until the nonproliferation assessment becomes publicly available. Some suggested that the nonproliferation 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 fuels 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 that were considered to be within the scope of the EIS are addressed in the Draft EIS.

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 have been added that treat blanket and driver 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 is conducting a separate cost study and a nuclear nonproliferation assessment for the reasonable alternatives. In response to public comment, completion of these reports has been expedited so that they are available to the public at the same time as the Draft EIS.

With respect to comments related to the ongoing electrometallurgical demonstration project, data from the project were used for the preparation of the Draft EIS. DOE expects that the National Research Council will issue a final report on the waste forms generated by the technology demonstration upon completion of the project in August 1999. DOE will consider the Council's final report in preparing the Final EIS and in the Record of Decision process which will follow.

S.1.2.2 Scope of This EIS

The EIS evaluates the potential direct, indirect, and cumulative environmental impacts associated with reasonable alternatives for 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. The 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 for the proposed action, as discussed in Section S.5.3.

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.

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, some of the alternatives under the proposed action do involve the separation of plutonium and highly enriched uranium. 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 will independently evaluate the impacts of each treatment technology on U.S. nonproliferation efforts. The nonproliferation assessment report will be published at about the same time as the Draft EIS.

S.1.2.3 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 should it 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 (e.g., GMODS, plasma arc) or new treatment technology.

The information presented in this EIS, combined with public comments on the Draft EIS, the nonproliferation assessment report, a separate 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.

S.1.2.4 Relationship to Other Actions and Programs

This section explains the relationship between this EIS and other relevant NEPA documents. Completed NEPA actions are described in Section S.1.2.4.1; ongoing actions are described in Section S.1.2.4.2.

S.1.2.4.1 Completed NEPA Actions

Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement (DOE/EIS-0203, April 1995)

The Programmatic Spent Nuclear Fuel EIS 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 Idaho National Engineering and Environmental Laboratory). 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 INEEL, this document analyzed alternatives such as no action, a 10-year plan, minimum and maximum treatment, storage, and disposal of DOE wastes. The SBSNF EIS is being prepared as a follow-on to this programmatic EIS.

Savannah River Site Waste Management Final Environmental Impact Statement (DOE/EIS-0217, October 1995)

DOE issued this EIS to provide a basis for the selection of a site-wide approach to managing present and future (through 2024) wastes generated at SRS. These wastes would come from ongoing operations and potential actions, new missions, environmental restoration, and decontamination and decommissioning programs. The SRS Waste Management EIS is relevant to the SBSNF EIS because it evaluates management alternatives for various types of waste that actions proposed in this SBSNF EIS could generate.

Final Environmental Impact Statement, Interim Management of Nuclear Materials (DOE/EIS-0220, December 1995)

In this EIS, 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. This Interim Management EIS evaluates treatment and management alternatives for spent nuclear fuel and other waste materials at SRS such as those generated by the proposed actions in the SBSNF EIS.

Environmental Assessment for the Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West (DOE/EA-1148, May 1996)

This NEPA analysis addressed the environmental impacts associated with a research and demonstration project involving the electrometallurgical treatment of up to 100 EBR-II driver assemblies and up to 25 EBR-II blanket 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 would treat the remainder of EBR-II spent nuclear fuel and/or other types of spent nuclear fuel.

Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement (DOE/EIS-0240, June 1996)

DOE prepared this EIS 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 Disposition of Surplus Highly Enriched Uranium EIS evaluates management alternatives for materials that actions proposed in this SBSNF EIS could generate.

Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste (DOE/EIS-0200, May 1997)

This Waste Management Programmatic EIS examined the potential environmental and cost impacts of strategic management alternatives for managing five types of radioactive and hazardous wastes 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 Waste Management 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 Waste Management Programmatic EIS evaluates management and treatment alternatives for various types of waste material that actions proposed in the SBSNF EIS could generate.

Advanced Mixed Waste Treatment Project Final Environmental Impact Statement (DOE/EIS-0290, January 1999)

This Advanced Mixed Waste EIS assessed the potential environmental impacts associated with four alternatives related to the construction and operation of a proposed Advanced Mixed Waste Treatment Facility at INEEL. The proposed Advanced Mixed Waste Treatment Facility would treat transuranic waste, alpha-contaminated mixed waste, and 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 Waste Management Programmatic EIS. This Advanced Mixed Waste EIS evaluates management and treatment alternatives for waste materials that actions proposed in the SBSNF EIS could generate.

S.1.2.4.2 Ongoing NEPA Actions

Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement (DOE/EIS-0229D, December 1998)

This SRS Spent Nuclear Fuel EIS analyzes the potential impacts from the management of spent nuclear fuel and targets assigned to SRS, including the placing of these materials in forms suitable for ultimate disposition. Options to treat, package, and store spent nuclear fuel are discussed in this document. The alternatives considered in the SRS Spent Nuclear Fuel EIS encompass a range of new packaging, new processing, and conventional reprocessing technologies for the treatment of spent nuclear fuel. Most of these reprocessing technologies are also analyzed in this SBSNF EIS. The SRS Spent Nuclear Fuel EIS was issued in December 1998. The Notice of Availability was published in the *Federal Register* on December 24, 1998 (63 FR 71285). The SRS Spent Nuclear Fuel EIS evaluates management and treatment alternatives for spent nuclear fuel and other waste materials that actions proposed in the SBSNF EIS could process and generate.

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

This document is in preparation. DOE is assessing 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 Yucca Mountain EIS will evaluate three thermal loading implementation alternatives: (1) high thermal load, (2) intermediate thermal load, and (3) low thermal load. The EIS will evaluate the environmental impacts of surface and below-ground construction, operation, and eventual closure activities, as well as national and regional transportation and various packaging options for shipping spent nuclear fuel and the high-level radioactive waste to the repository. The SBSNF EIS considers the potential disposal at Yucca Mountain of spent nuclear fuel or high-level radioactive waste that may result from the proposed action involving sodium-bonded spent nuclear fuel.

Idaho National Engineering and Environmental Laboratory High-Level Waste Environmental Impact Statement

This document is in preparation. DOE is preparing this EIS to evaluate alternatives for managing the high-level radioactive waste and associated radioactive wastes and facilities at INEEL. Under the terms of the 1995 Settlement Agreement/Consent Order with the State of Idaho, DOE agreed to treat high-level radioactive wastes currently stored at INEEL and to prepare the wastes 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 these radioactive wastes 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. This EIS evaluates management and treatment alternatives for waste types that actions proposed in the SBSNF EIS could generate.

S.2 SODIUM-BONDED SPENT NUCLEAR FUEL CHARACTERISTICS

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 of 2,500 metric tons of heavy metal.

The bulk of the 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 usually placed at the outer perimeter of the core and is used to breed plutonium-239, a fissile material. Blanket fuel primarily contains the nonfissile isotope uranium-238, which converts to fissile plutonium-239 as it absorbs the neutrons produced from the fission process. Typical blanket and driver spent nuclear fuel elements are shown schematically in **Figure S–1**.

The blanket and driver spent nuclear fuels addressed by this EIS contain metallic sodium between the cladding (outer layer of the fuel element) and the metallic fuel pins to improve heat transfer from the fuel to the reactor coolant through the stainless steel cladding. When driver fuel is irradiated for some period of time, the metallic fuel swells as fission products are generated until it reaches the cladding wall. During this process, metallic sodium enters the metallic fuel and becomes inseparable from it. In addition, fuel and cladding components interdiffuse to such an extent that mechanical stripping of the driver spent nuclear fuel cladding is not practical. On the other hand, when blanket fuel is irradiated, the metallic fuel does not swell to the same degree as the driver fuel because the burnup in the blanket fuel is low. As a result, minimal metallic sodium enters the fuel pin and there is no interdiffusion between fuel and cladding. This allows mechanical stripping of the blanket spent nuclear fuel cladding.

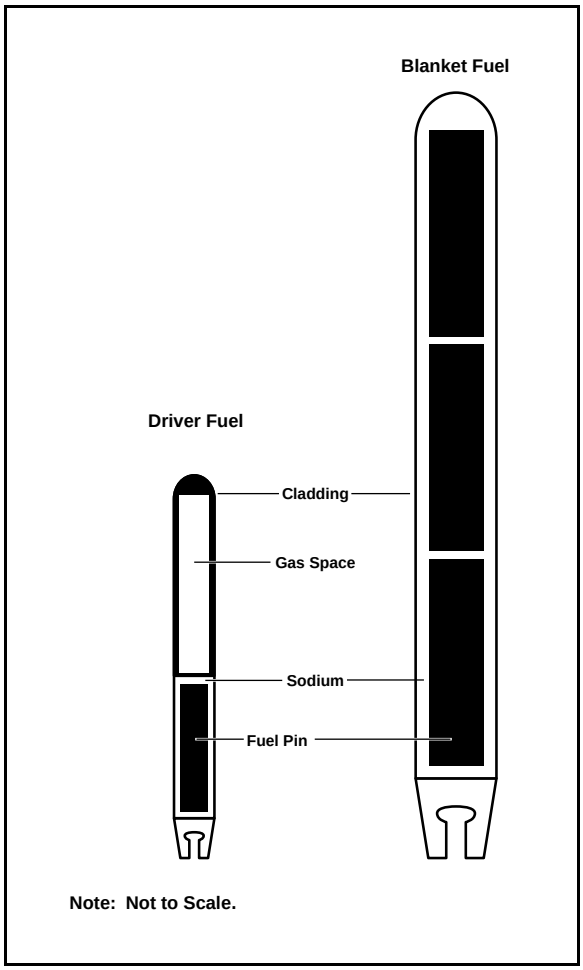


Figure S–1 Typical Driver and Blanket Spent Nuclear Fuel Elements

S.2.1 Experimental Breeder Reactor-II (EBR-II) Spent Nuclear Fuel

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 that is designed to simulate fission products). The typical EBR-II driver spent nuclear fuel pin is a metal alloy of 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

Sources of Sodium-Bonded Spent Nuclear Fuel

Experimental Breeder Reactor-II *(EBR-II) is a research and test reactor located at ANL-W that was 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 its 33 years of operation, numerous fuel designs were tested in EBR-II, and EBR-II spent nuclear fuel contains both driver and blanket fuel.*

Enrico Fermi Atomic Power Plant *(Fermi-1) was built at Monroe, Michigan (30 miles southwest of Detroit), to demonstrate the feasibility of the 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 spent nuclear fuel from Fermi-1 is sodium-bonded.*

Fast Flux Test Facility Reactor*, located on the Hanford Site near Richland, Washington, is a 400-megawatt-thermal nuclear test reactor cooled by liquid sodium. It was built in 1978 to test plant equipment and fuel for the U.S. Government's liquid metal reactor development program. Although the Fast Flux Test Facility Reactor is not a breeder reactor, this program demonstrated the technology of commercial breeder reactors. The sodium-bonded spent nuclear from the Fast Flux Test Facility Reactor is driver fuel.*

74-centimeter-long (29-inch-long) stainless steel tube (cladding) and welded shut. This unit of fuel is called an element. Sixty-one (91 in some fuels) 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 EBR-II blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in metal form. 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 operation of EBR-II. Blanket assemblies are similar to driver assemblies, except the blanket pins are made entirely from depleted uranium and the individual blanket pins are larger.

S.2.2 Fermi-1 Spent Nuclear Fuel

The Fermi-1 blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in a uranium-molybdenum alloy. 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.

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, and measure 3.4 meters, 6.4 centimeters (11 feet, 2.5 inches) long and 64.8 centimeters (25.5 inches) in diameter. The 14 canisters were placed into CPP-749, an underground dry storage system.

The total quantity of Fermi-1 blanket material is 34 metric tons of heavy metal. The blanket assemblies have a very low irradiation history, so the inventory of fission products, activation products, and transuranics is low.

S.2.3 Fast Flux Test Facility and Other Experimental Sodium-Bonded Spent Nuclear Fuel

DOE's inventory of sodium-bonded spent nuclear fuel includes eight liquid metal reactor test assemblies containing driver fuel that were irradiated at the Fast Flux Test Facility at Hanford. It also includes small quantities of fuel from liquid metal reactor experiments that have metallic sodium or an alloy of sodium and potassium. These miscellaneous small-lot fuels differ in cladding composition, uranium content, enrichment, and burnup. Some of the fuels consist of uranium and/or plutonium carbides and oxides in addition to metal uranium or alloy. They are stored at several DOE sites such as the Hanford Site, Oak Ridge, SRS, Sandia National Laboratories-New Mexico, and INEEL. Those lots stored outside INEEL will be transported to INEEL pursuant to the Record of Decision (60 FR 28680) for the Programmatic Spent Nuclear Fuel EIS.

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

	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	176
Fermi-1 Blanket	19	34	365
Fast Flux Test Facility Driver	8[b]	0.3	7
Miscellaneous	3[b]	0.1	31
Total	101	60	662

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

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

Table S–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*
EBR-II blanket	INEEL (ANL-W)	Radioactive Scrap and Waste Facility	Elements in canisters
		Fuel Conditioning Facility	In process material*
EBR-II driver	INEEL (INTEC)	CPP-603 pool	About 12 elements per canister
		CPP-666 pool	
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	Tech Area V	Experimental capsules
	SRS	Receiving Basin for Offsite Fuels	Elements
	Oak Ridge National Laboratory	Building 3525	Elements

* Being processed as part of the EBR-II electrometallurgical demonstration project.

S.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 ultimate disposal in a geologic repository. These are: the electrometallurgical treatment; 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. Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository without treatment, i.e., packaging the fuel in high-integrity cans with minimal preparation (cleaning and conditioning) without sodium removal, is not ruled out and has been considered in this EIS under the No Action Alternative.

S.3.1 Electrometallurgical Treatment Process

The electrometallurgical treatment process was developed at the Argonne National Laboratory for processing EBR-II blanket and driver spent nuclear fuel assemblies containing metallic fuel. The process has been demonstrated for the stainless steel-clad uranium alloy fuel used in that reactor. Modifications to the process could be used for the treatment of oxide and carbide sodium-bonded spent nuclear fuel. The electrometallurgical treatment process uses electrorefining, an industrial technology used to produce pure metals from impure metal feedstock. Electrorefining has been used to purify metal for more than 100 years.

The first step in processing sodium-bonded spent nuclear metallic fuel would be removal of the fuel elements from the fuel assemblies. The fuel elements then would be chopped into short segments and placed in stainless steel baskets in the electrorefiner, where the electrometallurgical treatment would occur. The electrorefiner would be maintained at high temperatures and would contain a molten mixture of primarily two salts. The chopped fuel elements would be lowered into the molten salt. Upon application of an electric voltage, the 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. The stainless steel cladding hulls and some of the insoluble fission products would remain in the baskets.

After a sufficient amount of spent nuclear fuel has been processed, the salt would be removed and solidified. The salt would be ground to a desired size and mixed with zeolite, a filter and ion-exchange agent, to collect certain fission products from 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 powder then would be added to the zeolite mixture, which would be hot-pressed to produce a ceramic high-level radioactive waste form that is expected to be suitable for ultimate disposal.

The uranium would be removed and treated to remove any adhered salts. Then it would be melted (and depleted uranium would be 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 ultimate disposal.

In addition to the ceramic and metal waste forms of high-level radioactive waste, some low-level radioactive waste would be generated.

S.3.2 Plutonium-Uranium Extraction (PUREX) Process

The PUREX process 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. The process 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. Use of these facilities

for treating sodium-bonded spent nuclear fuel involves certain restrictions inherent in the design: (1) the sodium complicates the process as employed in the SRS facilities; (2) the stainless steel cladding would require significant modifications or additions to the existing facilities; and (3) the presence of alloys (e.g., zirconium) 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 F-Canyon facility could be used without modifications for the blanket sodium-bonded spent nuclear fuel, if the spent nuclear fuel were declad and the sodium were removed prior to the process.

The fuel pins would be dissolved in an aqueous solution of nitric acid. The resulting solution containing uranium, plutonium, and fission products would undergo feed clarification and acidity/alkalinity adjustment. The clarified solution then would be treated via the PUREX process 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 streams 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 would be processed to a borosilicate glass form. Material streams from the PUREX process would be uranium trioxide, plutonium metal, and high-level radioactive waste.

S.3.3 High-Integrity Cans

High-integrity can packaging provides substitute cladding for damaged or declad fuel, or another level of containment for intact fuel. The can is constructed of a highly corrosion-resistant material to provide long-term protection in a repository. The high-integrity cans are placed into standardized canisters that are ready for disposal in waste packages. High-integrity cans could be used to store the sodium-bonded spent nuclear fuel on site until it can be shipped to a repository.

Packaging sodium-bonded spent nuclear fuel in high-integrity cans can be done with or without decladding and/or sodium removal. However, since the identified reason for the potential treatment of sodium-bonded fuel is the presence of metallic sodium, this method of packaging under the proposed action would require removal of the metallic sodium. Since sodium removal prior to treatment is not practical for driver spent nuclear fuel, this treatment method is applicable for blanket spent nuclear fuel after the metallic sodium has been removed. Packaging sodium-bonded spent nuclear fuel in high-integrity cans without sodium removal is considered in this EIS as a direct disposal option under the No Action Alternative.

The high-integrity cans would be placed in dry storage at ANL-W. Prior to placement in a repository, the high-integrity cans would be placed into a standardized canister designed to promote containment under repository conditions.

S.3.4 Melt and Dilute Process

The melt and dilute process involves chopping and melting the spent nuclear fuel and dilution by adding depleted uranium or other metals. There are three options for the melt and dilute process that are applicable to sodium-bonded spent nuclear fuel. 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 ANL-W using metallurgical technology developed for uranium and stainless steel cladding. In the second option, blanket spent nuclear fuel elements would be melted with cladding and additional stainless steel. In the first two options, there would be no actual dilution of the fissile component of the uranium because it is present in amounts far less than in natural uranium. The third option would involve developing a new melt and dilute process capable of handling sodium volatilized from processing the chopped driver spent nuclear fuel elements

with sodium and the cladding intact. In this process option, the fuel and stainless steel would be melted under a layer of material such as molten salt.

Under the first option, declad and cleaned blanket pins would be received at SRS in aluminum canisters, each containing some 60 kilograms of material. The canisters would be stored until they fit into the processing schedule. Following some validation of contents, the canisters would be loaded into a melting furnace with additional aluminum, if necessary. The furnace would operate at a very high temperature, 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.

Under the second option, blanket elements with the sodium removed would be loaded into a furnace crucible. A small amount of radioactive waste steel might be added to the crucible. The furnace would be heated to extremely high temperatures 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.

Under the third option, some of the sodium in the driver spent nuclear fuel elements would be removed in a similar manner to the sodium from blanket spent nuclear fuel elements. A melt and dilute process would be developed for driver spent nuclear fuel still containing the cladding and some metallic sodium. 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 to oxidize the molten sodium. Depleted uranium would be added to reduce the enrichment. A small portion of radioactive waste steel would be added. This furnace would be operated at the same temperature as in Option 2. The sodium would have to react in the flux salt to protect the off-gas treatment filter banks. After the melt is mixed, a vacuum would be applied to complete volatilization of the salt, which would be condensed and partially reused. The salt would be stabilized in the ceramic waste form. The metal melt would be stirred and cast into an ingot, placed in a standardized canister, and stored. The process can be used for sodium-bonded metallic spent nuclear fuel. Uranium nitride, oxide, and carbide fuels cannot be treated because of their high melting points.

S.3.5 Glass Material Oxidation and Dissolution System (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 very high temperature. 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 process has the potential for treating both blanket and driver sodium-bonded spent nuclear fuel, pending a successful research and development demonstration project to deal with sodium and other factors. 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 treatment would produce radioactive off-gases. These gases would be filtered and treated.

S.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 extremely high temperatures. A direct-current plasma torch would supply the energy required. 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, and the mixture would be homogenized by the torch. When the spent nuclear fuel is melted and oxidized throughout the ceramic, the rotation would be slowed to allow the molten vitreous ceramic to pour out by gravity flow into canister molds. The process has the potential for treating both blanket and driver sodium-bonded spent nuclear fuel, pending a successful research and development demonstration project to deal with sodium and other factors.

Metallic fuels such as EBR-II fuel would require the addition of some ceramic material. Depleted uranium could be added to reduce the uranium-235 enrichment 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.

S.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 the 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) extremely high-temperature chlorination and conversion of metallic fuel and cladding to gaseous chloride compounds; (2) removal of the transuranic chlorides and most of the fission products in a molten zinc chloride bed at a high temperature; (3) condensation of the other chlorides (e.g., uranium hexachloride) in a series of fluidized beds and condensers at 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.

S.3.8 Direct Disposal

Direct disposal of sodium-bonded spent nuclear fuel is disposal with minimum treatment, such as cleaning and conditioning. The sodium-bonded spent nuclear fuel (driver and blanket) would be packaged in high-integrity containers without removing the metallic sodium. The high-integrity cans would be placed into a standardized canister designed to provide containment under repository conditions during pre-closure operations. At the present time, Resource Conservation and Recovery Act (RCRA) mixed waste (which contains both hazardous and radioactive waste) does not meet the requirements of acceptable waste as identified in the DOE Civilian Radioactive Waste Management Office's March 19, 1999, draft *Waste Acceptance System Requirements Document* (DOE 1996). Because it contains metallic sodium, the sodium-bonded spent nuclear fuel could be categorized as a RCRA hazardous waste that is potentially both pyrophoric and reactive. 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.

S.3.9 Sodium Removal and Disposition

As discussed in 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. Direct disposal also does not call for sodium removal, so the metallic sodium would be a constituent in the disposal package.

For the methods that require the removal of sodium prior to treatment and/or decladding of the 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.

To remove the cladding after sodium has been extracted, a special machine would be installed. This machine would mechanically push the fuel pins within the cladding out through the opening created when the cladding ends of the fuel elements were previously cut off. Experience with unirradiated blanket spent nuclear fuel at ANL has shown that the pins could be mechanically pushed out of the stainless steel cladding after all the sodium bond 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. The remaining sodium would be removed during the process as nonreactive salt, stabilized in a ceramic waste form, and disposed of as low-level radioactive waste.

Table S–3 provides a summary for sodium removal and sodium disposition for the treatment methods addressed in this EIS.

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

Treatment and Management Methods	*Decladding Required*	*Sodium Treatment*	*Sodium Disposition*
Electrometallurgical Process Blanket and Driver Fuel	No	Stabilization	Converted into nonreactive form, as part of the process, and disposed of with the high-level ceramic radioactive waste product of the process.
High-Integrity Cans Blanket Fuel	No	Removal	Converted into nonreactive form, separate from the process, and disposed of as low-level radioactive waste.
PUREX Process Blanket Fuel	Yes	Removal	Converted into nonreactive form, separate from the process, and disposed of as low-level radioactive waste.
Melt and Dilute Process Driver Fuel	No	Removal	Converted into nonreactive form, separate 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 low-level radioactive waste.
Blanket Fuel	Yes	Removal	Converted into nonreactive form, separate from the process, and disposed of as low-level radioactive waste.
Direct Disposal Blanket and Driver Fuel	No	No	Disposed of in metallic reactive form in high-integrity cans.

S.4 SPENT NUCLEAR FUEL TREATMENT AND MANAGEMENT FACILITIES

For each alternative, DOE would require the use of 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

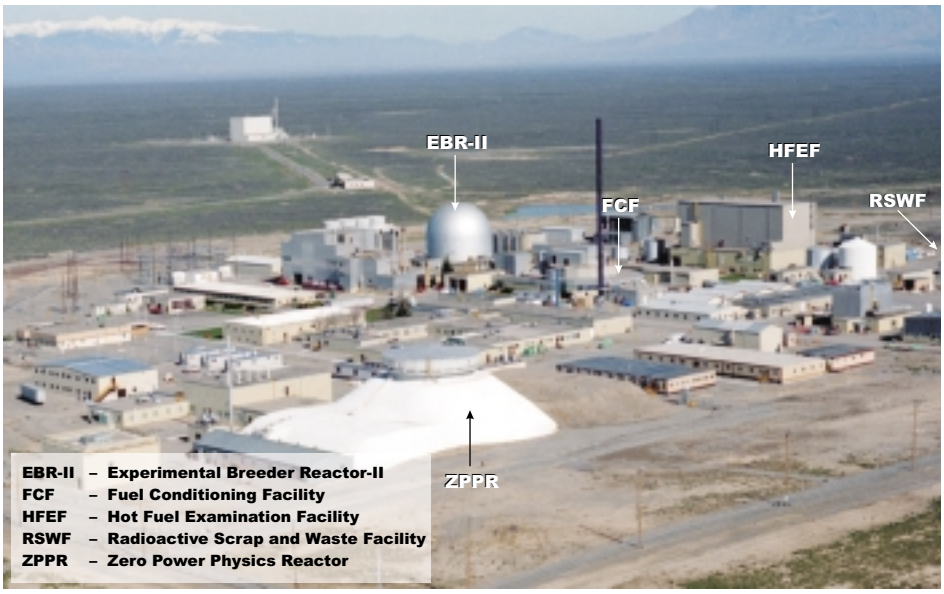
vacuum-dry the spent nuclear fuel. These facilities also could be used to repackage the fuel into storage canisters and place the repackaged fuel in dry interim storage to await treatment.

S.4.1 Argonne National Laboratory-West

The ANL-W site is a center of nuclear technology development and testing. The location of ANL-W is shown in **Figure S–2**. 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.

The Fuel Conditioning Facility is one of the facilities proposed for use in treating and managing the 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. The hot cells enable technicians to work safely with radioactive nuclear materials from behind 1.5-meter-thick (5-foot-thick) shielding walls. The air cell is used for handling, storage, and assembly/disassembly of components. The argon cell is a much larger, doughnut-shaped hot cell where personnel can work from the outside corridor around the hot cell and work in the hot cell can be monitored from an inner shielded work space in the center of the hot cell.

The Hot Fuel Examination Facility is also proposed for use in treating and managing 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 to support a wide variety of programs and process demonstrations. A wide range of remote operations and examinations may be performed in this facility with its shielded cells, support areas, and equipment. 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. 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



Argonne National Laboratory-West

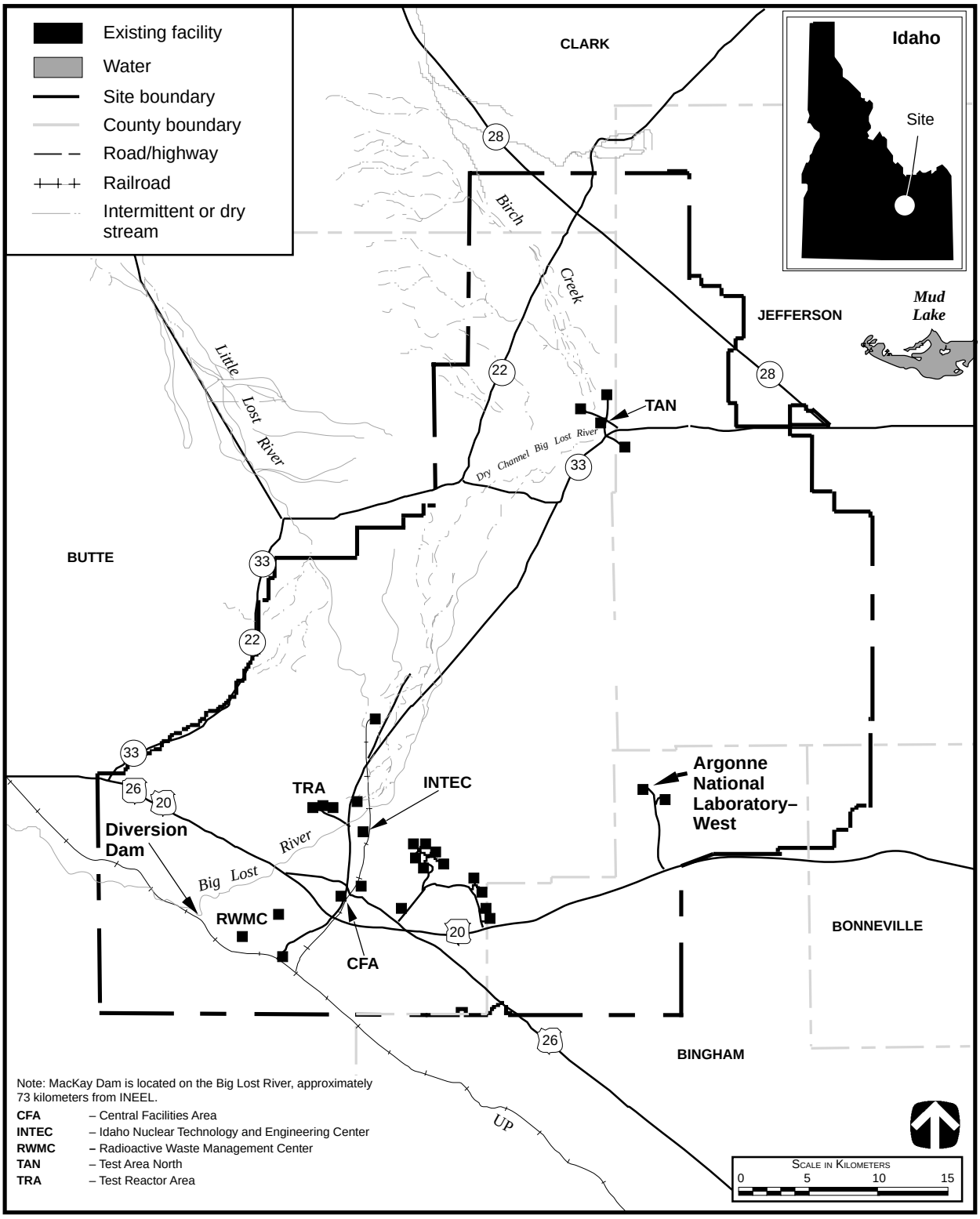


Figure S–2 Map of Idaho National Engineering and Environmental Laboratory

truck lock is large enough to accommodate the various trucks and fork lifts that transport the shielded casks used in the day-to-day operation of the facility. The facility recently was modified to accept truck-sized spent nuclear fuel shipping casks.

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

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

The Idaho Nuclear Technology and Engineering Center (INTEC, formerly known as the Idaho Chemical Processing Plant) is located northeast of the Central Facilities Area at INEEL. It is one of the sites where the sodium-bonded spent nuclear fuel is currently stored. INTEC was constructed in the 1950s to reprocess spent nuclear fuel from government reactors. In 1992, DOE announced it no longer would reprocess spent nuclear fuel. Current work at INTEC includes receiving and storing spent nuclear fuel, solidifying liquid high-level radioactive waste, environmental restoration and decontamination and dismantling activities, and technology development. For the proposed action, the facility would be used to package the currently stored sodium-bonded spent nuclear fuel for direct disposal if treatment is not necessary. However, because it has no hot cell with an inert gas atmosphere, INTEC cannot be used for any sodium removal activities under the proposed action.

S.4.2 Savannah River Site

The SRS (shown in **Figure S–3**) was constructed during the early 1950s to produce the basic materials used to fabricate nuclear weapons, primarily tritium and plutonium-239. The five reactors built on the site produced nuclear materials by irradiating target materials with neutrons. In addition, several support facilities were constructed on the site, including two chemical separations plants, a heavy water extraction plant, a nuclear fuel and target fabrication facility, and waste management facilities. As a result of changing defense requirements, all five of the original SRS production reactors have been 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.

Historically, irradiated materials were moved from the SRS reactors to the two chemical separations facilities–the next step in the production process. In these facilities, known as "canyons," the irradiated fuel and target assemblies were chemically processed to separate useful products from waste. The F-Canyon at SRS could be used to chemically separate uranium from fission products in blanket spent nuclear fuel using the PUREX process. DOE uses the F-Canyon chemical separation facility and the FB-Line to stabilize spent nuclear fuel and to recycle plutonium scrap generated from facility operations and offsite sources. In September 1997, the FB-Line began a new plutonium packaging process that places stabilized plutonium in rugged, welded stainless steel cans. DOE has determined the FB-Line should be used to stabilize the plutonium recovered from spent nuclear fuel. This current program will require the FB-Line to operate until about 2002.

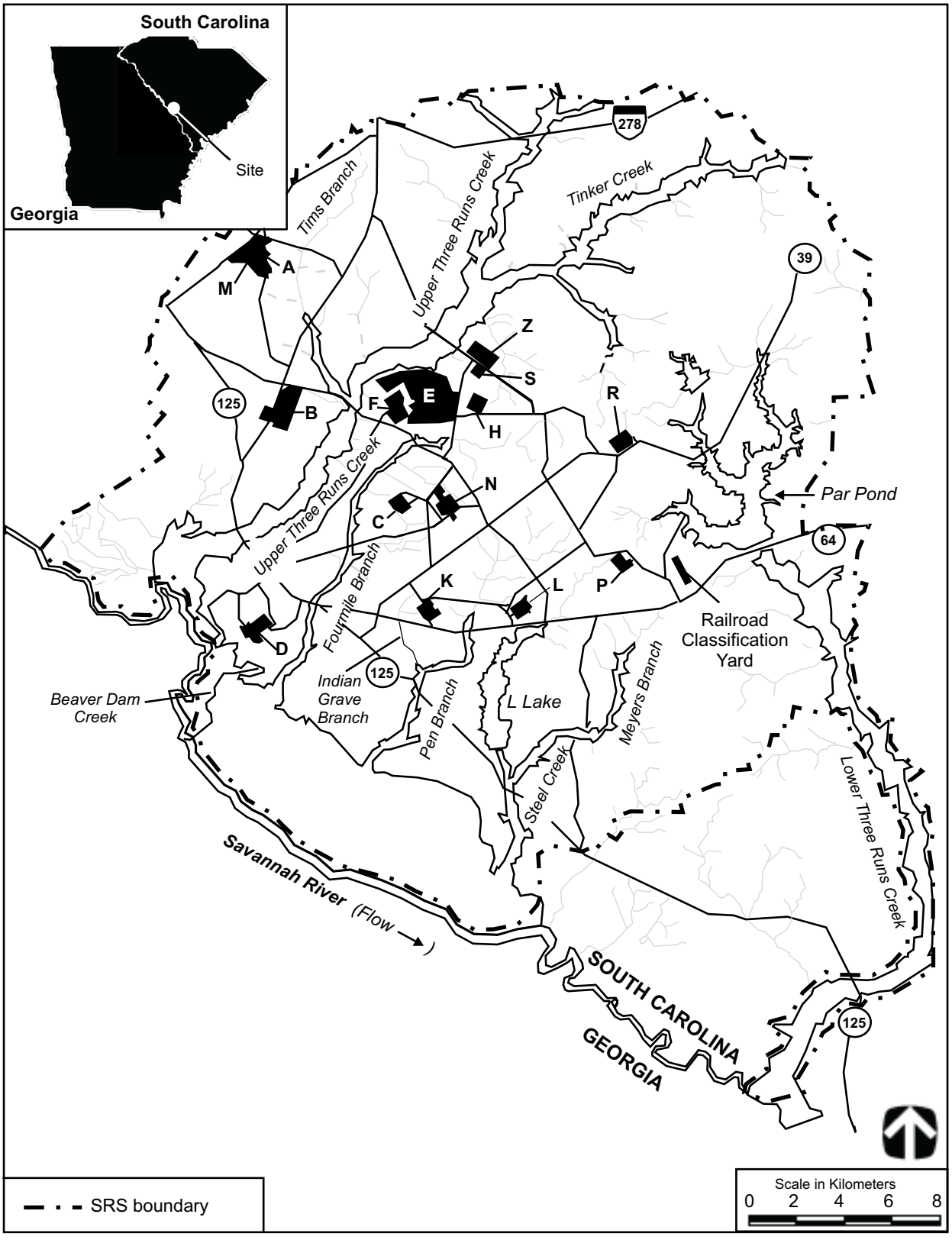


Figure S–3 Map of Savannah River Site

F-Canyon at SRS

In the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement*, DOE identified melt and dilute as one of the preferred methods for treating spent nuclear fuel at SRS. Building 105-L, part of the shutdown L-Reactor complex, is the SRS facility where installation of a melt and dilute process for treating spent nuclear fuel is proposed. 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.

To implement the melt and dilute technology, DOE would construct a melt and dilute facility in Building 105-L and build a dry-storage facility in L Area, near Building 105-L. DOE expects the melt and dilute option would be relatively simple to implement in Building 105-L. The major technical issue for 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 *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement*.

Building 105-L at SRS

The Defense Waste Processing Facility is another SRS facility that potentially could be used to treat sodium-bonded spent nuclear fuel. This facility currently is being used to convert high-level radioactive liquid waste stored at SRS into a solid borosilicate glass form that is suitable for long-term storage and disposal.

S.5 DESCRIPTION OF ALTERNATIVES

The proposed action is to treat and manage sodium-bonded spent nuclear fuel. The alternatives under the proposed action are illustrated in **Figure S–4** and are addressed below. Although each alternative addresses both blanket and driver spent nuclear fuel, DOE will consider the blanket and driver fuels separately in identifying a preferred alternative and any subsequent record of decision. In other words, DOE will consider all combinations of technologies, options and fuel types, including combinations not among the specific combinations explicitly considered 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.

S.5.1 No Action Alternative

Under the No Action Alternative, 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 are analyzed: (1) the sodium-bonded spent nuclear fuel would continue to be stored indefinitely at its current location in accordance with the Record of Decision (60 FR 28680) for the Programmatic Spent Nuclear Fuel EIS (DOE 1995) and other existing site-specific NEPA documentation or until another technology, currently dismissed as a reasonable alternative because of immaturity (i.e., 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, e.g., the fuel would be packaged in high-integrity cans with minimal preparation (cleaning and conditioning) and without sodium removal.

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 to treat both blanket and driver spent nuclear fuels, and require minimal preconditioning of the sodium-bonded 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 S.5.3.

S.5.2 Alternative 1: Electrometallurgical Treatment of Blanket and Driver Fuel at ANL-W

Under this alternative, the sodium-bonded blanket and driver spent nuclear fuel (approximately 60 metric tons of heavy metal) from ANL-W's Radioactive Scrap and Waste Facility and 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 spent fuel would be separated from the subassembly 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.

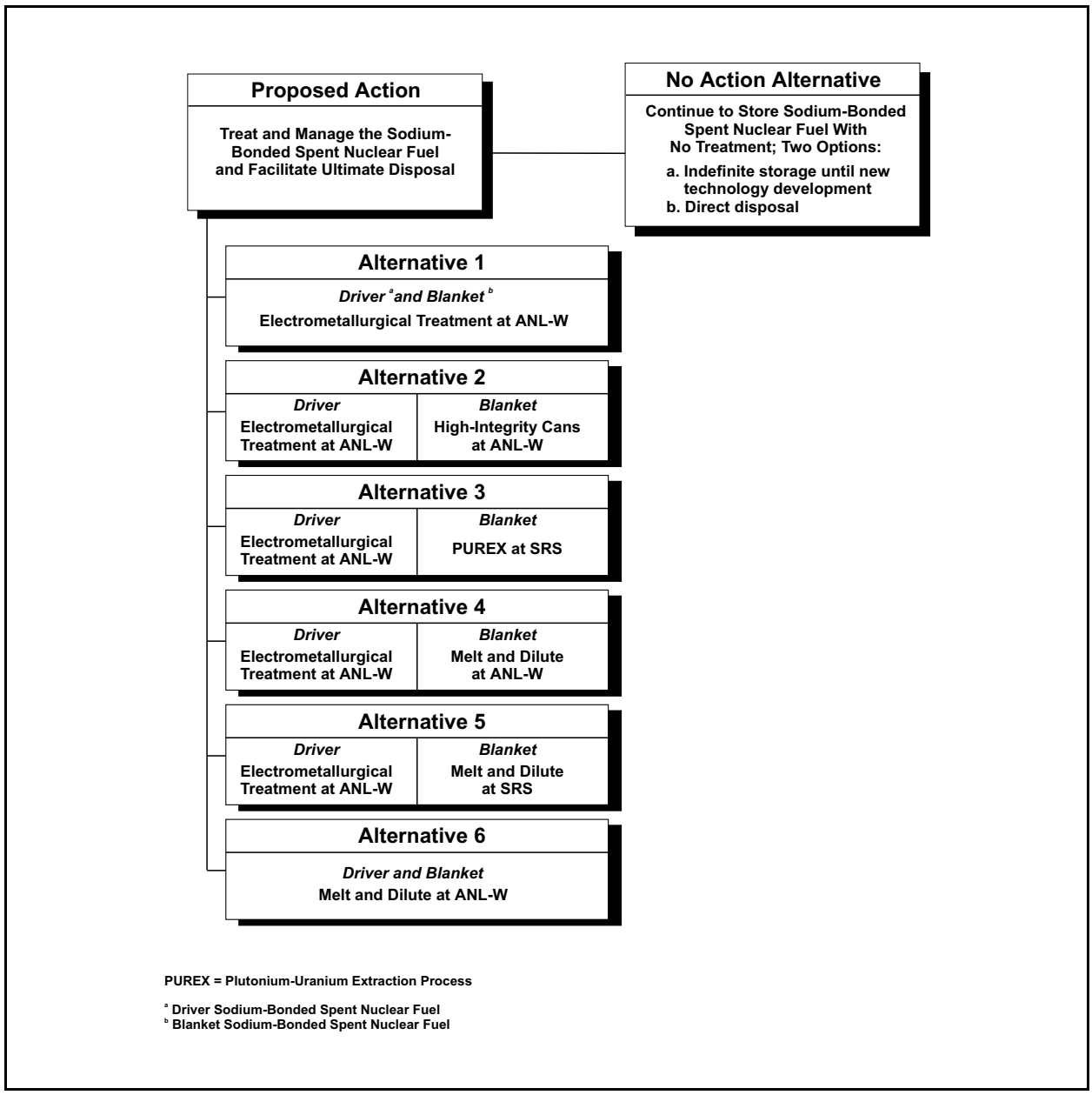


Figure S–4 Proposed Action and Alternatives

After treatment, the low-enriched uranium byproduct would be metal-cast at the Fuel Conditioning Facility and transferred to the Zero Power Physics Reactor Materials Storage Building for storage. The remaining cladding hulls would be packaged and transferred to the Hot Fuel Examination Facility for metal casting into high-level radioactive waste and would be transferred afterwards to the Radioactive Scrap and Waste Facility for storage. The electrorefiner salt containing the fission products, sodium, and transuranic elements would be transferred in metallic 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 12 to 13 years to process all fuel. Driver spent nuclear fuel alone would require approximately seven years.

S.5.3 Alternative 2: Remove Sodium and Package Blanket Fuel in High-Integrity Cans and Treat (Electrometallurgical Treatment) 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 removal of the sodium without decladding, as discussed in Section S.3.9. The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal, excluding approximately 0.08 metric tons of heavy metal in carbide fuels) would be treated using the electrometallurgical treatment process described in Section S.5.2.1 (Alternative 1).

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.

Implementing this alternative at either the Fuel Conditioning Facility or the Hot Fuel Examination Facility would require the installation of equipment for sodium removal activities. No new equipment would be needed for the electrometallurgical treatment of the driver sodium-bonded spent nuclear fuel under this alternative.

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.

S.5.4 Alternative 3: Declad and Clean Blanket Fuel and Treat (Electrometallurgical Treatment) 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 S.3.9.

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 S.5.2.1 for Alternative 1. The decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal could 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. Separated plutonium in metal form would be stored in an SRS vault. Depleted uranium would be transferred to the Defense Waste Processing Facility for storage.

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. 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.

S.5.5 Alternative 4: Melt and Dilute Blanket Fuel and Treat (Electrometallurgical Treatment) 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 the facility at ANL-W using the melt and dilute Option 2 process described in Section S.3.4. Prior to treatment, the metallic sodium would be removed without decladding at ANL-W, as discussed in Section S.3.9.

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 S.5.2.1 for Alternative 1.

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 at the facility. Equipment necessary for the melt and dilute process also would need to be installed at the facility, including the addition of the melter and an off-gas system.

Metal 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.

Treatment of the blanket spent nuclear fuel at ANL-W using the melt and dilute process could start as early as 2005 and could be completed in seven years. Treatment of the driver spent nuclear fuel could start as early as 2000 and could be completed in approximately seven years.

S.5.6 Alternative 5: Declad and Clean Blanket Fuel and Treat (Electrometallurgical Treatment) 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 the 105-L Building at SRS and treated using the melt and dilute Option 1 process, as described in Section S.3.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal), excluding approximately 0.08 metric tons of heavy metal in carbide fuels, would be treated at ANL-W using the electrometallurgical treatment process described in Section S.5.2.1 for Alternative 1.

Decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal would take place at the Hot Fuel Examination Facility at ANL-W. After decladding and sodium removal, the blanket spent nuclear fuel rods 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 the 105-L Building, 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 *Savannah River Site Spent Nuclear Fuel Environmental Impact Statement*.

Metal 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.

Treatment of the driver spent nuclear fuel at ANL-W could start in 2000 and could be completed in approximately seven years. Treatment of the blanket spent nuclear fuel at SRS would start around 2035. The

facility would be operational in 2005 and already 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.

S.5.7 Alternative 6: Melt and Dilute Driver and Blanket 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 S.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.

Removal of the sodium from the blanket 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 metal 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.

The melt and dilute process at ANL-W could start as early as 2005 and would take approximately 10 years to be completed for all blanket and driver spent nuclear fuels.

S.5.8 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 S.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 aluminum-clad spent nuclear fuel at SRS (DOE 1998b). Use of the melt and dilute process for sodium-bonded driver spent nuclear fuel only requires technology enhancement that DOE already has proposed for treating other spent nuclear fuels. In addition, unlike the other technologies that would require no new construction, both of these technologies 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.

Glass Material Oxidation and Dissolution System Process

The GMODS process, although similar to the melt and dilute process because of its thermal treatment, has not been developed beyond laboratory scale. Several developmental steps would be required before it can be deemed a mature process. These include: detailed process development, resolution of containment concerns, testing, and pilot plant demonstration to address technology risks (for example, 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 would be required to treat the radioactive elements volatilized at 1,000 °C (1,832 °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 plasma arc-vitreous ceramic process is being used for the vitrification of mixed wastes. 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 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 risks, 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 the 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 Cladded Blanket Spent Nuclear Fuel Using SRS PUREX Process

As discussed in Section S.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 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.

S.5.9 Preferred Alternative

When the Notice of Intent to prepare this EIS was published in the *Federal Register* on February 22, 1999 (64 FR 8553), the proposed action was the electrometallurgical treatment of DOE's inventory of sodium-bonded spent nuclear fuel in the Fuel Conditioning Facility at ANL-W. In response to public comments received during the scoping process, DOE reformulated the scope of the EIS to address more generally the treatment and management of DOE sodium-bonded spent nuclear fuel and to separate the technical analysis of blanket and driver spent nuclear fuel. Under the revised proposed action, several technology alternatives, including various combinations for blanket and driver spent nuclear fuel, have been analyzed in this Draft EIS. Information developed in the course of preparing this EIS suggests that alternative technologies may have certain advantages (e.g., cost) for some or all fuel. Accordingly, DOE has no preferred alternative at this time. DOE will consider the environmental analyses in this EIS, the public comments, and the findings of the independent cost study and the nonproliferation report, as well as other program policy factors, in determining a preferred alternative in the Final EIS.

S.6 Affected Environment

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. It is located primarily within Butte County, but portions of the site are also in Bingham, Jefferson, Bonneville, and Clark counties. Much of INEEL is open space that has not been designated for specific use. Land use at INEEL includes facility operations, grazing, general open space, and infrastructure (such as roads). The site lies in a cool desert ecosystem dominated by shrub-steppe vegetative communities. Developed portions of INEEL occur within the 93,000-hectare (230,000-acre) central core area of the site. ANL-W is located in the southeast portion of the central core area, about 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. Other than internal modification to existing facilities, no new construction would take place within ANL-W for any of the proposed alternatives.

SRS is located on about 80,130 hectares (198,000 acres) in southwest South Carolina. The site is 40 kilometers (25 miles) southeast of Augusta, Georgia, and 19 kilometers (12 miles) south of Aiken, South Carolina. It is bordered by the Savannah River to the southwest and includes portions of three South Carolina counties: Aiken, Allendale, and Barnwell. Land use at SRS includes forest and undeveloped areas, water and wetlands, and developed facilities. Land use in F-Area is classified as heavy industrial, with facilities that historically have been associated with chemical and physical processes used to separate uranium, plutonium, and fission products. Land use in L-Area also is classified as heavy industrial, with facilities that historically

have been associated with nuclear materials production for national defense. Other than internal modification to existing facilities, no new construction would take place within SRS for any of the proposed alternatives.

S.7 CONSEQUENCES

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 in this EIS (see Section 2.5 of this EIS). The information presented in this section is based on Chapter 4, which provides a detailed discussion of the impacts on the 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 in detail, DOE has determined that they would have minimal or no impacts on the remaining environmental areas (e.g., land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources) at the proposed sites. This is because the proposed facilities already exist so, except for internal building modifications and new equipment installation, no construction activities would be required.

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 S–4**.

S.7.1 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 safe, secure, indefinite storage at current locations, 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 with minimal preparation.

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 indefinite 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. The impacts summarized below would be applicable to both options considered under the No Action Alternative.

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. 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 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 623 indirect jobs. 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 only risk to the health and safety of the 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 under the No Action Alternative at ANL-W and INTEC would be small. The annual dose to the population within 80 kilometers (50 miles) from these releases would be

Radiological Health Effects Risk Factors Used in this EIS

Health impacts of radiation exposure, whether from sources external or internal to the body, are generally identified as "somatic" (i.e., affecting the exposed individual), or "genetic" (i.e., affecting descendants of the exposed individual). Radiation is more likely to produce somatic effects than genetic effects. Except for leukemia, which can have an induction period (time between exposure to carcinogen and cancer diagnosis) of as little as 2 to 7 years, most cancers have an induction period of more than 20 years.

For a uniform irradiation of the body, the incidence of cancer varies among organs and tissues; the thyroid gland and skin demonstrate a greater sensitivity than other organs. Such cancers, however, also produce comparatively low mortality rates because they are relatively amenable to medical treatment. Because fatal cancer is the most probable serious effect of environmental and occupational radiation exposure, estimates of cancer fatalities, rather than cancer incidence, are presented in this EIS. The numbers of cancer fatalities can be used to compare the risks of various alternatives.

Risk factors are used to calculate the statistically expected effects of exposing a population to ionizing radiation. For example, in a population of 100,000 people exposed only to natural background radiation (300 millirem per year), about 15 latent cancer fatalities per year would be expected (100,000 persons x 0.3 rem per year x 0.0005 latent cancer fatalities per person-rem = 15 latent cancer fatalities per year).

The number of latent cancer fatalities corresponding to a single individual's exposure over a presumed 72-year lifetime to 0.3 rem per year is 0.011 (1 person x 0.3 rem per year x 72 years x 0.0005 latent cancer fatality per person-rem = 0.011 latent cancer fatality). Presented another way, this method estimates that approximately 1.1 percent of the population might die of cancers induced by background radiation. The same calculations apply to workers with one difference: the risk factor for workers is 0.0004 latent cancer fatalities per person-rem instead of 0.0005 cancer fatalities per person-rem for the general public.

The health consequences of exposure to radiation from normal operation and accidents are converted to estimates of cancer fatality risks using dose conversion factors recommended by the International Commission on Radiological Protection. For individuals, the EIS provides estimates of probability of a latent cancer fatality occurring for the involved and noninvolved workers, the maximally exposed individual, and an average individual in the general population. These categories are defined as follows:

Involved Worker: *An individual actively participating in the proposed action.*

Noninvolved Worker: *An individual 640 meters (0.4 miles) from the radioactive material release point.*

Maximally Exposed Offsite Individual: *A hypothetical individual who could potentially receive the maximum dose of radiation or hazardous chemicals.*

General Population: *Individuals within an 80-kilometer (50-mile) radius of the facility.*

0.022 person-rem per year. The risk that an average individual would develop a fatal cancer from this exposure would be 0.000011, or a probability of one cancer fatality in 90,900 years. 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 0.00077 millirem per year, and the risk of developing a fatal cancer from this exposure would be 3.9×10^{-10} (once in 2.5 billion years). The average worker would receive 60 millirem per year, and the risk of developing a cancer from this exposure would be 0.000024, or once in 41,666 years.

The risk of developing a fatal cancer from postulated accident conditions associated with either option under the No Action Alternative at ANL-W would be 6.9×10^{-8} per rem for the population within 80 kilometers (50 miles), or one in 17 million years. Since INTEC is further away from the INEEL site boundary and major population centers compared to ANL-W, the accident impacts would be less than those presented for ANL-W. The annual cancer risk for the maximally exposed offsite individual would be 6×10^{-10} (one in two billion years), and for the noninvolved worker it would be 1.9×10^{-10} or one in five billion years.

Hazardous Chemical Exposures

Hazardous chemical impacts resulting from either option under 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 values, indicate that under either option of the No Action Alternative the worst postulated accident conditions would result in no adverse health effects to a worker or the maximally exposed offsite individual.

Environmental Justice

As discussed above, the impacts from either option under 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 and INTEC. These include low-level radioactive, transuranic, mixed, hazardous, and nonhazardous wastes. They are associated with the operation of the facilities where the sodium-bonded spent nuclear fuel is stored. High-level radioactive waste in metal and ceramic forms generated as a result of completing waste processing of the electrometallurgical treatment demonstration project 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 wastes are presented in Table S–4.

Transportation

No offsite transportation activities would occur under either option of the No Action Alternative.

S.7.2 Proposed Action Impacts

Under the proposed action, the EIS evaluates six distinct alternatives, as described in Section S.5 and illustrated in Figure S-4. Alternative 1 proposes to treat both the blanket and driver 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 blanket and driver spent nuclear fuel using the melt and dilute method at ANL-W.

All alternatives under the proposed action have very small impacts on air quality, water resources, socioeconomics, public and occupational health and safety, and transportation areas of the environment in and around the INEEL/ANL-W and SRS locations. For all alternative, the radiological and nonradiological gaseous emissions and liquid effluents, as well as the associated exposures to workers and the public, are 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 consideration in this EIS, the volume of high-level radioactive waste for all the considered alternatives is an important consideration. All the proposed action alternatives result in a decrease in high-level radioactive waste volume as compared to the direct disposal No Action Alternative: 45 percent (Alternative 1); 70 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 will not change as a result of the proposed action.

Radiological gaseous emissions would be small and 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 per year of elemental tritium and 11,600 (Alternative 1) to 32,250 (Alternative 6) curies per year of krypton-85.

Radiological gaseous emissions at SRS would be 54 (Alternative 5) to 162 (Alternative 3) curies per year of elemental tritium and 399 (Alternative 5) to 1,188 (Alternative 3) curies per year 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.

No changes are 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 S–4 indicates some small quantities of tritium and other radionuclides, (including strontium, ruthenium, and isotopes of uranium and plutonium) would be released. 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 S–4, the risk is small for all alternatives considered under the proposed action.

Radiological Exposures

Comparing alternatives at ANL-W, the annual population dose from routine gaseous radioactive releases would range from 0.0029 person-rem (Alternative 1) to 0.012 person-rem (Alternative 6), with a latent cancer fatality risk in the range of 1.5×10^{-6} to 6.0×10^{-6}, respectively.

The 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 annual dose to an average individual within the 80-kilometer (50-mile) population would range from 0.000012 millirem (Alternative 1) to 0.00051 millirem (Alternative 6), with a latent cancer fatality risk in the range of 6.0×10^{-12} to 2.6×10^{-11}, respectively.

The collective dose to workers at ANL-W would be 22 person-rem for all alternatives. This corresponds to an 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.

Comparing alternatives at SRS, the maximum population dose from routine gaseous radioactive releases would range from 0.0076 person-rem per year (Alternative 5) to 0.02 person-rem for the whole treatment period (Alternative 3), corresponding to additional latent cancer fatalities in the range of 3.8×10^{-6} to 0.000010, respectively.

The maximum dose to the maximally exposed offsite individual would range from 0.00010 millirem per year (Alternative 5) to 0.00051 millirem (Alternative 3) for the whole treatment period, with a latent cancer fatality risk in the range of 5.0×10^{-11} to 2.6×10^{-10}, respectively. The dose to an average individual within the 80-kilometer (50-mile) population would range from 0.000011 millirem per year (Alternative 5) to 0.000024 millirem (Alternative 3), with a latent cancer fatality risk in the range of 5.3×10^{-12} to 1.2×10^{-11}, respectively.

The maximum collective dose to workers at SRS would be 50 person-rem per year (Alternative 5). This corresponds to additional latent cancer fatalities of 0.075. 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 0.00010 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 0.0088×10^{-3} (Alternative 6, driver spent nuclear fuel, design-basis earthquake). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 0.000076 (Alternative 6, driver spent nuclear fuel, design-basis earthquake). The highest annual latent cancer fatality risk for the noninvolved worker would be 2.7×10^{-6} (Alternative 6, driver spent nuclear 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 (Alternative 1, driver spent nuclear fuel, beyond-design-basis earthquake). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 2.2×10^{-7} (Alternative 1, driver spent nuclear fuel, beyond-design-basis earthquake). The highest annual latent cancer fatality risk for the noninvolved worker would be 1.7×10^{-9} (Alternative 1, blanket spent nuclear 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 0.013 (Alternative 5, blanket spent nuclear fuel, loss of cooling water). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 6.6×10^{-6} (Alternative 5, blanket spent nuclear fuel, loss of power). The highest annual latent cancer fatality risk for the noninvolved worker would be 3.1×10^{-6} (Alternative 5, blanket spent nuclear fuel, loss of power).

Hazardous Chemical Exposures

Hazardous chemical impacts from normal operation 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, if any.

Hazardous chemical impacts under accident conditions, evaluated in terms of comparison to Emergency Response Planning Guideline values, indicate that under the proposed action all postulated hazardous chemical releases would not result in long-term adverse health effects to a worker, or any adverse health effects to a maximally exposed offsite individual at either ANL-W or SRS.

Waste Management

Table S–4 presents a comparison of the volumes of high-level radioactive, low-level radioactive, and transuranic wastes generated by each of the alternatives. Each of the alternatives generates from 37 to 84 percent less high-level radioactive waste, compared to the No Action Alternative option of the direct disposal of spent nuclear fuel. Alternative 2 also generates less low-level radioactive waste when compared to the No Action Alternative. In comparison, all other alternatives would generate greater volumes of low-level radioactive waste. Each of the alternatives generates more transuranic waste, but Alternatives 1, 2, 4, and 6 would only exceed this waste volume by a range of 7 to 41 percent. Alternatives 3 and 5 generate significantly greater volumes of transuranic waste, between 2.3 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 generated by Alternative 3 for blanket spent nuclear fuel has been shown to meet current criteria. It is expected, however, that other waste forms (e.g., ceramic, metal, and high-integrity cans not containing metallic sodium) also would be suitable for repository disposal.

Transportation

The transportation activities under Alternatives 1, 2, 4, and 6 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 4.7×10^{-5} person-rem; the dose to the public would be 3.5×10^{-4} person-rem. Accordingly, incident-free transportation activities would result in 1.9×10^{-8} latent cancer fatalities among transportation workers and 1.7×10^{-7} latent cancer fatalities in the total affected population over the duration of the transportation activities.

The dose to the population from postulated accidents from these activities would be less than 1×10^{-12} person-rem, resulting in less than 1×10^{-15} latent cancer fatalities. Nonradiological traffic fatalities would be 8.2×10^{-7}.

Transportation activities under Alternatives 3 and 5 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 2×10^{-3} person-rem; the dose to the public would be 0.013 person-rem. Accordingly, incident-free transportation activities would result in 7.9×10^{-7} latent cancer fatalities among transportation workers and 6.1×10^{-6} latent cancer fatalities in the total affected population over the duration of the transportation activities. Nonradiological fatalities among the public from vehicle emissions during intersite transportation would be 1.96×10^{-4}.

The dose to the population from postulated accidents from these activities would be 3.0×10^{-6} person-rem, resulting in 1.5×10^{-9} latent cancer fatalities. Nonradiological traffic fatalities would be 0.002.

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Table S–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/year)	Negligible impact Tritium: 460[a] Krypton-85: 7,120[a]		Negligible impact Tritium: 770 Krypton-85: 11,600		Negligible impact Tritium: 809 Krypton-85: 11,860	
Water Resources - Radiological liquid effluents	No impact No liquid effluent		No impact No liquid effluent		No impact No liquid effluent	
Socioeconomics	Loss of 350 direct jobs and 623 indirect jobs; no measurable impact		Workforce maintained; No impact		Workforce maintained; No impact	
Public and Occupational Health and Safety						
• Project duration	35 years		13 years		9 years	
• Normal operation	***Person-rem/yr***[c]	***LCF***	***Person-rem/yr***	***LCF***	***Person-rem/yr***	***LCF***
- Population dose	0.022	0.000011	0.0029	1.5×10^{-6}	0.0031	1.6×10^{-6}
- MEI	0.00077	3.9×10^{-10}	0.00034	1.7×10^{-10}	0.00038	1.9×10^{-10}
- Average individual	0.000092	4.6×10^{-11}	0.000012	6.0×10^{-12}	0.000013	6.6×10^{-12}
- Total worker	22	0.0088	22	0.0088	22	0.0088
- Average worker	60	0.000024	60	0.000024	60	0.000024
Hazardous chemicals						
- MEI / worker (impacts)	None					
• Accidents Maximum annual cancer risk (per year)						
- 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]		1.5×10^{-8} (DBA); 1.5×10^{-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	
- Worker	Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
Environmental Justice	No disproportionately high and adverse impact to minority and low-income populations					
Waste Management (cubic meters)						
• High-level radioactive wastes	152 (Direct disposal SNF volume)[d]		84.3		43.9[e]	
• Low-level radioactive wastes	812		861		733.7	
• Transuranic wastes	10		14.1		10.7	
Transportation						
• Incident-free	***Person-rem***	***LCF***	***Person-rem***	***LCF***	***Person/rem***	***LCF***
- Population	No impact		0.00035	1.7×10^{-7}	0.00035	1.7×10^{-7}
- Workers			4.7×10^{-5}	1.9×10^{-8}	4.7×10^{-5}	1.9×10^{-8}
• Accidents						
- Population			1.1×10^{-12}	1.0×10^{-15}	1.1×10^{-12}	1.0×10^{-15}

Alternative 1: *Electrometallurgical Treatment of Blanket and Driver Spent Fuel at ANL-W*
Alternative 2: *Package Blanket Spent Fuel in High-Integrity Cans and Treat Driver Fuel at ANL-W*
Alternative 3: *Declad and Clean Blanket Spent Fuel and Treat Driver Spent Fuel at ANL-W; PUREX Process Blanket Fuel at SRS*
Alternative 4: *Melt and Dilute Blanket Spent Fuel and Treat Driver Spent Fuel at ANL-W*
Alternative 5: *Declad and Clean Blanket Spent Fuel and Treat Driver Spent Fuel at ANL-W; Melt and Dilute Blanket Fuel at SRS*
Alternative 6: *Melt and Dilute Blanket and Driver Spent Fuel at ANL-W*

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: 399		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	
Workforce maintained; no impact		Workforce maintained; no impact		Workforce maintained; no impact		Workforce maintained; no impact		Workforce maintained; no impact		Workforce maintained; no impact	
9 years		Less than 1 year		12 years		9 years		3 years		10 years	
Person-rem/yr	***LCF***	***Person-rem/yr***	***LCF***	***Person-rem/yr***	***LCF***	***Person-rem/yr***	***LCF***	***Person-rem/yr***	***LCF***	***Person-rem/yr***	***LCF***
0.0031	1.6×10^{-6}	0.02	0.000010	0.0031	1.6×10^{-6}	0.0031	1.5×10^{-6}	0.0076	3.8×10^{-6}	0.012	6.0×10^{-6}
0.00038	1.9×10^{-10}	0.00051	2.6×10^{-10}	0.00038	1.9×10^{-10}	0.00038	1.9×10^{-10}	0.00010	5.0×10^{-11}	0.002	1.0×10^{-9}
0.000013	6.6×10^{-12}	0.000024	1.2×10^{-12}	0.000013	6.6×10^{-12}	0.000013	6.6×10^{-12}	0.000011	5.5×10^{-12}	0.000051	2.6×10^{-11}
22	0.0088	38	0.015[a]	22	0.0088	22	0.0088	50	0.02	22	0.0088
60	0.000024	250	0.0001[a]	60	0.000024	60	0.000024	500	0.0002	60	0.000024
None	None	Small	Small	None	None	None	None	Small	Small	None	None
5.6×10^{-6} (DBA); 0.000013 (BDBA)		0.00017 (DBA)		0.00022 (DBA); 0.000013 (BDBA)		5.6×10^{-6} (DBA); 0.000013 (BDBA)		0.013		0.0088 (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)		4.8×10^{-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 and low-income populations											
24.3 (18.7 at ANL-W, 5.6 at SRS)				64.3		95.32 (18.7 at ANL-W, 76.62 at SRS)				86	
2,960.5 (770.5 at ANL-W, 2,190 at SRS)				828		1,178.5 (770.5 at ANL-W, 408 at SRS)				924	
100.7 (10.7 at ANL-W, 90 at SRS)				12.8		23.2 (6.7 at ANL-W, 16.5 at SRS)				14.1	
Person-rem	***LCF***			***Person-rem***	***LCF***	***Person-rem***	***LCF***			***Person-rem***	***LCF***
0.013	6.1×10^{-6}			0.00035	1.7×10^{-7}	0.013	6.1×10^{-6}			0.00035	1.7×10^{-7}
0.002	7.9×10^{-7}			4.7×10^{-5}	1.9×10^{-8}	0.002	7.9×10^{-7}			4.7×10^{-5}	1.9×10^{-8}
3.0×10^{-6}	1.5×10^{-9}			1.1×10^{-12}	1.0×10^{-15}	3.0×10^{-6}	1.5×10^{-9}			1.1×10^{-12}	1.0×10^{-15}

ERPG = Emergency Response Planning Guideline, LCF = latent cancer fatalities, MEI = maximally exposed individual
DBA = design-basis accident, BDBA = beyond-design-basis accident, SNF = spent nuclear fuel

[a] Only occurs for the initial two years during fuel packaging and handling, and Electrometallurgical Treatment Demonstration Project waste stabilization.

[b] Over a period of six months.

[c] Population doses (population and total worker) are in person-rem per year; individual doses are in millirem.

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

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

S.8 GLOSSARY

Background Radiation — Ionizing radiation present in the environment from cosmic rays and natural sources in the Earth; background radiation varies considerably with location.

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.

Borosilicate Waste Glass — Glass typically containing 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).

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, its purpose is 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.

Cladding — The outer jacket of fuel elements usually made of aluminum, stainless steel, or zirconium alloy, used to prevent fuel corrosion and retain fission products during reactor operation, or to prevent releases into the environment during storage.

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. The includes processing and passivation of spent nuclear fuel.

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.

Decladding — The process of mechanically removing the cladding from the fuel pin in a fuel element.

Degraded (spent nuclear fuel) — Spent nuclear fuel whose external cladding has cracked, pitted, corroded, or potentially allows the leakage of radioactive materials.

Depleted Uranium — Uranium with a smaller percentage of uranium-235 than the 0.7 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 it to another material.

Disposal — The isolation of radioactive wastes 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.

Disassembly — Removal of the fuel elements from the fuel assembly.

Dose — The energy imparted to matter by ionizing radiation. The unit of absorbed dose is the rad.

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.

Driver Fuel — Those fuel tubes or elements composed of enriched uranium, placed at the center of the reactor core, and used to sustain the fission chain reaction.

Effluent (liquid) — Wastewater, treated or untreated, that flows out of a treatment plant, sewer, or industrial outfall; generally refers to wastes discharged into surface waters.

Emission — A material discharged into the atmosphere from a source operation or activity.

Fission Products — Nuclei formed by the fission of heavy elements (primary fission products); also, the nuclei formed by the decay of the primary fission products, many of which are radioactive.

Fuel Assembly — A cluster of fuel elements (or rods). Approximately 200 fuel assemblies make up a reactor core.

Fuel Element — Nuclear reactor component that includes the fissile material (fuel pin) sealed and the cladding.

Fuel Pin —The uranium metal or alloy that undergoes fission in a nuclear reactor (without cladding).

Geologic Repository — A system that is intended to be used for, or may be used for, the disposal of radioactive waste and spent nuclear fuel in excavated geologic media. A geologic repository includes (a) the geologic repository operations area, and (b) the portion of the geologic setting that provides isolation. A near-surface disposal area is not a geologic repository.

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.

High-Level Radioactive Waste — The 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 other highly radioactive material that is determined, consistent with existing law, to require permanent isolation.

Latent Fatalities — Fatalities associated with acute and chronic environmental exposures to chemical or radiation that occur within 30 years of exposure.

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.

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 DOE site boundary nearest to the facility in question. A hypothetical person who could potentially receive the maximum dose of radiation or hazardous chemicals.

Metric Tons of Heavy Metal (MTHM) — 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.

Millirem — One thousandth of a rem.

Mixed Waste — Waste that contains both "hazardous waste" and "radioactive waste" as defined in this glossary.

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.

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.

Person-Rem — The unit of collective radiation dose to a given population; the sum of the individual doses received by a population segment.

Pyrophoric — Being highly susceptible to spontaneous ignition and continuous combustion.

Radioactive Waste — Materials from nuclear operations that are radioactive or are contaminated with radioactive materials, and for which use, reuse, or recovery are impractical.

Reprocessing (of spent nuclear fuel) — Processing of reactor-irradiated nuclear material (primarily spent nuclear fuel) to recover fissile and fertile material, in order to recycle such materials primarily for defense programs. Historically, reprocessing has involved aqueous chemical separations of elements (typically uranium or plutonium) from undesired elements in the fuel.

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.

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.

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. It is not a mixed waste.

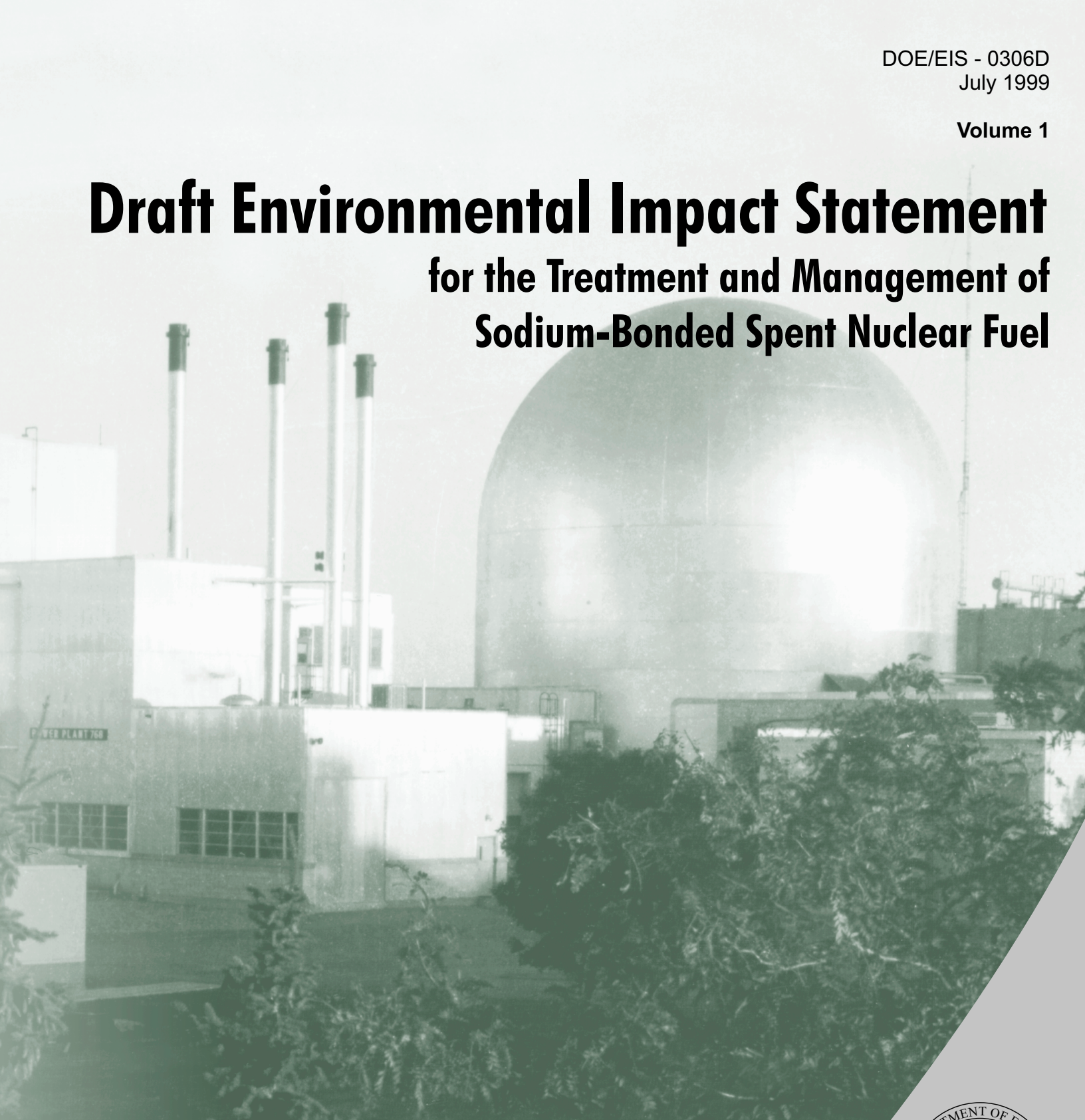
Treatment — In this EIS, a process to remove and/or stabilize metallic sodium.

DOE/EIS - 0306D
July 1999

Volume 1

Draft Environmental Impact Statement

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



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

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Abstract: The Department of Energy (DOE) is responsible for the safe and efficient management of several different types of spent nuclear fuel. One type of spent nuclear fuel that may not be suitable for disposal in a geologic repository without treatment is the DOE-owned sodium-bonded spent nuclear fuel. Sodium-bonded spent nuclear 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 material could complicate the process of qualifying and licensing such spent nuclear fuel for disposal in a geologic repository. Currently, more than 98 percent of DOE's sodium-bonded spent nuclear fuel is located at the Idaho National Engineering and Environmental Laboratory (INEEL). In a 1995 agreement with the State of Idaho, DOE committed to remove all spent nuclear fuel from Idaho by 2035.

Several technologies for spent nuclear fuel treatment are under development and might facilitate qualification and licensing for ultimate disposal. The most developed technology is the electrometallurgical treatment of sodium-bonded spent nuclear fuel at Argonne National Laboratory-West (ANL-W). This EIS evaluates the potential environmental impacts associated with the treatment of sodium-bonded spent nuclear fuel in one or more spent nuclear fuel management facilities: ANL-W at INEEL (near Idaho Falls, Idaho) and either the F-Canyon or Building 105-L at the Savannah River Site (near Aiken, South Carolina). The EIS analyzes under the proposed action the electrometallurgical process, the plutonium-uranium extraction (PUREX) process, direct disposal in high-integrity cans with the sodium removed, and the melt and dilute process. The EIS also evaluates the continued storage of sodium-bonded spent nuclear fuel and direct disposal without treatment under the No Action Alternative.

Public Comments: In preparing this Draft EIS, DOE considered comments received from the public during the scoping process (February 22, 1999 to April 8, 1999). Comments on this Draft EIS may be submitted during the 45-day comment period. Public meetings on this EIS will also be held during the comment period. The dates, times, and locations of these meetings will be announced shortly after issuance of this Draft EIS.

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ACRONYMS, ABBREVIATIONS, AND CONVERSION CHARTS

ANL	Argonne National Laboratory
ANL-W	Argonne National Laboratory-West
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
EMT	Electrometallurgical Treatment (of spent fuel)
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
IAEA	International Atomic Energy Agency
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
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
P.L.	Public Law
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

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 5/9ths	Celsius	Celsius	Multiply by 9/5ths, 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 period. 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) near Idaho Falls, Idaho; the Enrico Fermi Atomic Power Plant at 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, which is 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 facilitate 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 1998a), the final acceptance criteria will be more refined. If the repository is developed, final acceptance criteria would not be available until after the U.S. Nuclear Regulatory Commission (NRC) issues its construction authorization, based on the successful demonstration of the safe, long-term performance of the repository in accordance with NRC regulations. Until such time, the preliminary acceptance criteria will tend 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 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) 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. This Record of Decision was partially based on the conclusions of the Programmatic Spent Nuclear Fuel EIS (DOE 1995a), 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. It also analyzed the consequences of 10 years of waste and spent nuclear fuel management and environmental restoration actions at Idaho National Engineering Laboratory.[1]

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, involves the treatment of 100 EBR-II driver assemblies and up to 25 EBR-II blanket assemblies (approximately 1.6 metric tons of heavy metal). The driver fuel contains highly enriched uranium and was used in the active region of the nuclear reactor core. The blanket fuel contains depleted uranium and was used in areas around and near the driver fuel in the reactor core. The electrometallurgical treatment research and demonstration project is scheduled to be completed in August 1999. After completing the demonstration project, DOE will need to take further action to prepare the rest of the sodium-bonded fuel for disposal.

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 ongoing demonstration project were used in preparing this Draft SBSNF EIS. The National Research Council will issue a final report on the waste forms generated by the technology demonstration after the August 1999 completion of the project. DOE will consider the Council's final report in preparing the Final EIS and reaching a decision regarding the disposition of sodium-bonded spent nuclear fuel.

[1] *The laboratory's name was changed 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 complicate the disposal certification and licensing for the ultimate disposal of this spent nuclear fuel in a geologic repository. Metallic sodium reacts vigorously with water or moist air, 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 which used these fuels. To the extent possible, sodium was removed from the external surfaces of these fuels after their 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 repository acceptance criteria probably will exclude reactive materials unless their packaging minimizes the probability of rapid oxidation (DOE 1998b). Finally, some of the sodium-bonded spent nuclear fuel contains highly enriched uranium, and its disposal in a geologic repository may require special criticality control measures.

The presence of reactive or pyrophoric materials such as metallic sodium and metallic uranium, or the presence of highly enriched uranium, could complicate the process of certification of the spent nuclear fuel for disposal. Such qualification would require sufficient data and predictive analyses to demonstrate that emplacement of the spent nuclear fuel would not adversely affect a repository's ability to protect the environment and worker and public health and safety.

To ensure that the State of Idaho Settlement Agreement is 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 fuels. 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 to promote consistency between the two efforts and to minimize programmatic risks associated with waste form qualification and acceptance for ultimate disposal.

1.3 ISSUES IDENTIFIED DURING 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.

Approximately 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 raised during the public scoping period are addressed below.

Many commentors at the public 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)
Plutonium-uranium extraction (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 Independent Assessment Final Report on the demonstration project, a nonproliferation assessment report, and an independent cost study. Several commentors said that this EIS is premature because the electrometallurgical treatment 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 assessment report in the same time frame as the Draft EIS, or that this EIS should be delayed until the nonproliferation assessment becomes publicly available. Some suggested that the nonproliferation 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 fuels 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 Draft 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 Argonne National Laboratory-West 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 have been added that treat driver and blanket fuel by different technologies.

Issues related to cost and nuclear nonproliferation were not considered to be within the scope of the EIS. However, DOE is conducting a separate cost study and a nuclear nonproliferation assessment for the reasonable alternatives. In response to public comment, completion of these reports has been expedited so they are available to the public in the same time frame as the Draft EIS.

With respect to comments related to the ongoing electrometallurgical demonstration project, data from the project were used for the preparation of the Draft EIS. DOE expects that the National Research Council will issue a final report on the waste forms generated by the technology demonstration upon completion of the project in August 1999. DOE will consider the Council's final report in preparing the Final EIS and in the Record of Decision process which will follow.

The 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.

1.4 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. The 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 for 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, one of the alternatives under the proposed action involves the separation of plutonium and highly enriched uranium. 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 will independently evaluate the impacts of each treatment technology on U.S. nonproliferation efforts. The nonproliferation assessment report will be 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 assessment report, a separate 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 National Environmental Policy Act (NEPA) documents. Completed NEPA actions are described in Section 1.6.1; ongoing actions are described in Section 1.6.2.

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 Idaho National Engineering and Environmental Laboratory). 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 Idaho National Engineering and Environmental Laboratory, this document analyzed alternatives such as no action, a 10-year plan, and minimum and maximum treatment, storage, and disposal of DOE wastes.

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 Idaho National Engineering and Environmental Laboratory; (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 is being 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 between DOE, the State of Idaho, and the 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) wastes generated at SRS. These wastes 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, 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 wastes 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- and H-Canyons 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 assemblies and up to 25 EBR-II blanket assemblies in the Fuel Conditioning Facility at ANL-West. 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 would 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.

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 wastes 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 two 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 in New Mexico, which will transfer its transuranic waste to the Los Alamos National Laboratory in New Mexico. 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.

1.6.1.7 Advanced Mixed Waste Treatment Project Final Environmental Impact Statement

This EIS (DOE 1999) assessed the potential environmental impacts associated with four alternatives related to the construction and operation of a proposed 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 proposed Advanced Mixed Waste Treatment Facility would treat transuranic waste, alpha-contaminated mixed waste, and 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 would treat and prepare for shipment and disposal 65,000 cubic meters (2.30 million cubic feet) of DOE transuranic waste, alpha-contaminated mixed waste, and mixed wastes 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 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 Environmental Impact Statement 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.

1.6.2 Ongoing NEPA Actions

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

The *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement* was issued in December 1998. The Notice of Availability was published in the *Federal Register* on December 24, 1998 (63 FR 71285). This draft SRS EIS (DOE 1998b) analyzes the potential impacts from the management of spent nuclear fuel and targets assigned to SRS, including the placing of 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 EIS encompass a range of new packaging, new processing, and conventional reprocessing technologies for the treatment of spent nuclear fuel. Many of these technologies are also analyzed in this SBSNF EIS. However, in the SRS 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 Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada, Environmental Impact Statement

This document is in preparation. DOE is assessing 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 EIS will evaluate three thermal-loading implementation alternatives: (1) high thermal load, (2) intermediate thermal load, and (3) low thermal load. The EIS will evaluate the environmental impacts of surface and below-ground construction, operation, and eventual closure activities, as well as national and regional transportation and various packaging options for shipping spent nuclear fuel and high-level radioactive waste to the repository. The SBSNF EIS considers the potential disposal at Yucca Mountain of spent nuclear fuel or high-level radioactive waste that may result from the proposed action involving sodium-bonded spent nuclear fuel.

1.6.2.3 Idaho National Engineering and Environmental Laboratory High-Level Radioactive Waste and Facilities Disposition Environmental Impact Statement

This document is in preparation. DOE is preparing this EIS to evaluate alternatives for managing the high-level radioactive waste and associated radioactive wastes and facilities at INEEL. Under the terms of the 1995 Settlement Agreement/Consent Order with the State of Idaho, DOE agreed to treat high-level radioactive wastes currently stored at INEEL and to prepare the wastes 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 these radioactive wastes 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. This EIS evaluates treatment alternatives for wastes that actions proposed in the SBSNF EIS could generate.

In this EIS, DOE evaluates reasonable alternatives and options for the treatment of high-level radioactive waste, sodium-bearing wastes, newly generated wastes, 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 (INTEC).

1.7 ORGANIZATION OF THE EIS

This EIS volume contains 8 chapters and 11 appendices. The main analyses are included in the chapters and additional project information is provided in the appendices. The 8 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 period; 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; and background information on the ultimate disposition of spent nuclear fuel

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 index

The 11 appendices contain the following information: public scoping process 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; and a contractor disclosure statement.

1.8 REFERENCES

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NAS (National Academy of Sciences), 1995, *An Assessment of Continued R&D into an Electrometallurgical Approach for Treating DOE Spent Nuclear Fuel*, Committee on Electrometallurgical Techniques for DOE Spent Fuel Treatment, Washington, DC.

NAS (National Academy of Sciences), 1998, *Electrometallurgical Techniques for U.S. Department of Energy Spent Fuel Treatment-Spring 1998 Status Report on Argonne National Laboratory's R&D Activity*, Committee on Electrometallurgical Techniques for DOE Spent Fuel Treatment, Washington, DC.

2. PROPOSED ACTION AND ALTERNATIVES

Chapter 2 describes the proposed action and alternatives to treat and manage the sodium-bonded spent nuclear fuel. 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 in this environmental impact statement. 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 reasonable alternatives.

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 the 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 the reasonable alternatives, the following sections provide background information on the characteristics, inventory, and current storage locations of the sodium-bonded spent nuclear fuel; 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 fuels 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 various sodium-bonded spent nuclear fuels in DOE's inventory are included in Appendix D.

Proposed Action

DOE proposes to treat and manage the sodium-bonded spent nuclear fuel and facilitate ultimate 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 usually placed at the perimeter of the core and is used to breed the fissile material plutonium-239. It primarily contains 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 0.10 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 fuel elements are shown schematically in **Figure 2–1**.

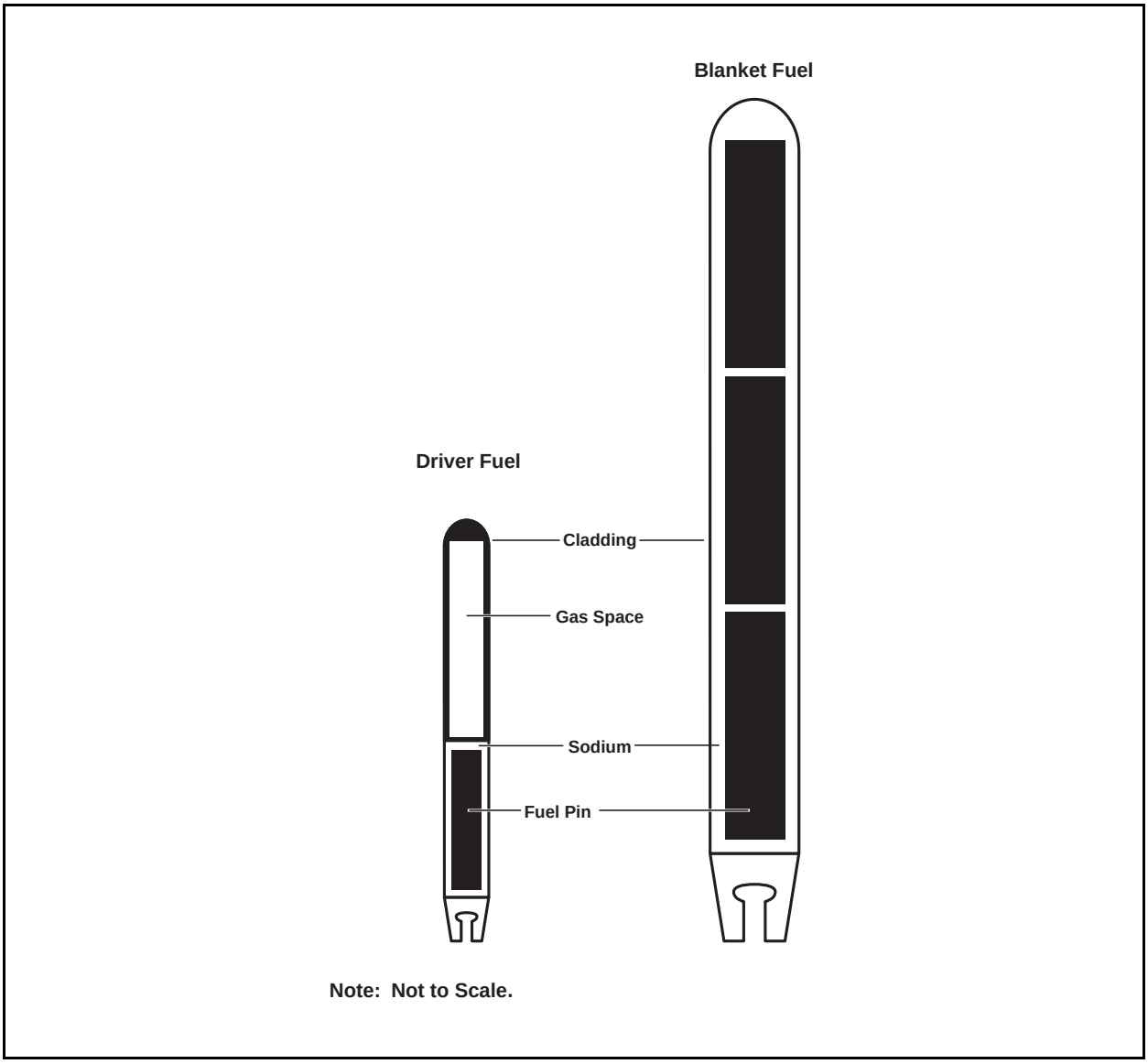

Figure 2–1 Typical Driver and Blanket Spent Nuclear Fuel Elements

The blanket and driver spent nuclear fuels addressed in this 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 is also 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 be expected. 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 Experimental Breeder Reactor-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 designed to simulate fission products). 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 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 in some fuels) 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 metal 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. 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

The Enrico Fermi Atomic Power Plant[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.

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.

[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 (EF-1).*

2.2.3 Fast Flux Test Facility and Other Experimental 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 fuels consist of uranium and/or plutonium carbides and oxides in addition to metal uranium or alloy. They are located at several DOE sites such as the Hanford Site, Oak Ridge, Savannah River Site (SRS), Sandia National Laboratories/New Mexico, and the Idaho National Engineering and Environmental Laboratory (INEEL). Those outside INEEL will be transported to INEEL pursuant to the Record of Decision (60 FR 28680) for the Programmatic Spent Nuclear Fuel EIS (DOE 1995a). Under the proposed action, it is assumed that they will be stored at ANL-W.

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

	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	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.

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.

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 ultimate 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. Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository without treatment, i.e., packaging the fuel in high-integrity cans with minimal preparation (cleaning and conditioning) without sodium removal, is not ruled out and has been considered in this EIS under the No Action Alternative.

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*
EBR-II blanket	INEEL (ANL-W)	Radioactive Scrap and Waste Facility	Elements in canisters
		Fuel Conditioning Facility	In process material*
EBR-II driver	INEEL (INTEC)	CPP-603 pool	About 12 elements per canister
		CPP-666 pool	
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	Tech Area V	Experimental capsule
	SRS	Receiving basin for offsite fuels	Element
	Oak Ridge National Laboratory	Building 3525	Elements

* Being processed as part of the EBR-II electrometallurgical demonstration project.

2.4 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 ultimate 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. Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository without treatment, i.e., packaging the fuel in high-integrity cans with minimal preparation (cleaning and conditioning) without sodium removal, is not ruled out and has been considered in this EIS under the No Action Alternative.

2.4.1 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 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.

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.

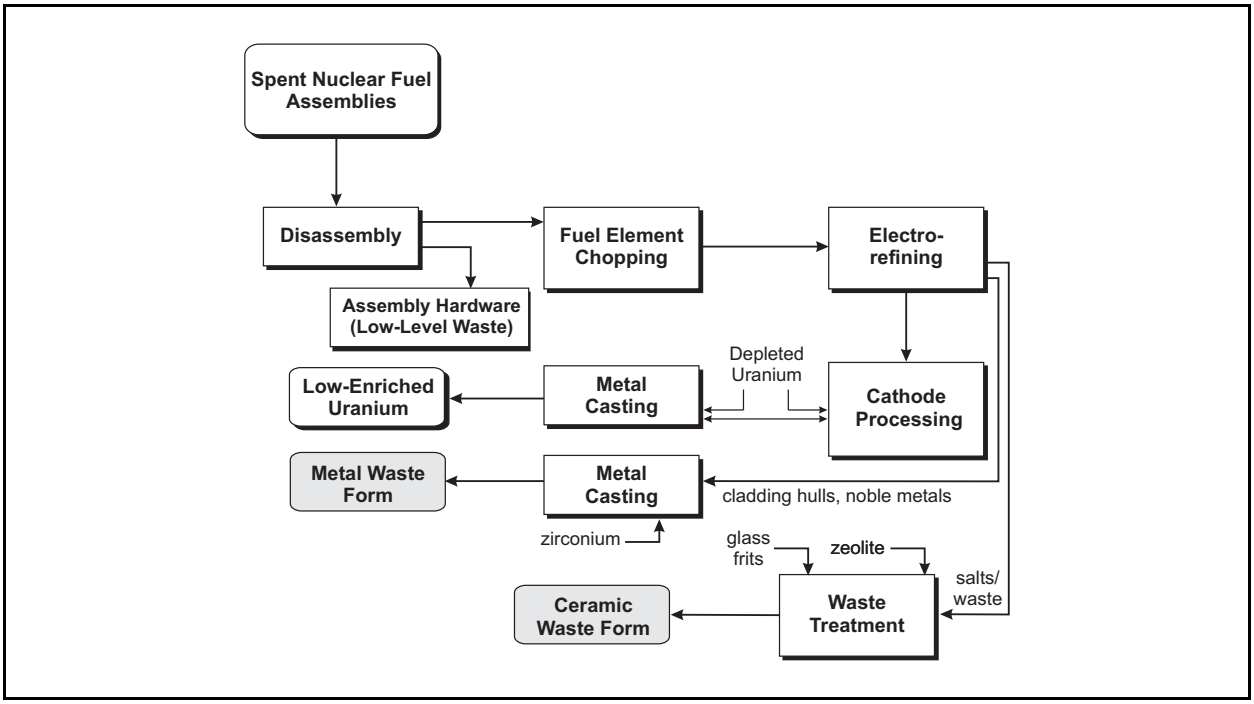

Figure 2–2 Electrometallurgical Treatment Process Flow Diagram

The electrorefiner, in which the electrometallurgical treatment occurs, would be maintained at 500 °C (932 °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 powder 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 ultimate 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 ultimate disposal.

The oxide fuels 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 fuels and blanket elements and 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 dilute 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.

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 subassemblies in the form of hardware. A detailed description of the electrometallurgical treatment process is presented in Appendix C.

2.4.2 Plutonium-Uranium Extraction (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 trioxide, plutonium metal, and high-level radioactive waste. **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.

2.4.3 High-Integrity Cans

The high-integrity can packaging provides substitute cladding for damaged or declad fuel, or another level of containment for intact fuel. The can is constructed of a highly corrosion-resistant material (Hastelloy Alloy C-22) to provide for long-term corrosion protection in a repository environment. The can could be used to store fuel on site until it is ready to be shipped to the repository. The high-integrity cans are placed into standardized canisters ready for disposal in waste packages.

Packaging sodium-bonded spent nuclear fuel in high-integrity cans can be done with or without decladding and/or sodium removal. However, since the identified reason for the potential treatment of sodium-bonded fuel is the presence of metallic sodium, this method of packaging under the proposed action would require the removal of the metallic sodium. Since sodium removal prior to treatment is not practical for driver sodium-

bonded spent nuclear fuel, this treatment method is applicable for blanket spent nuclear fuel after the metallic sodium has been removed. Packaging sodium-bonded spent nuclear fuel in high-integrity cans without sodium removal is analyzed in this EIS as a direct disposal option under the No Action Alternative.

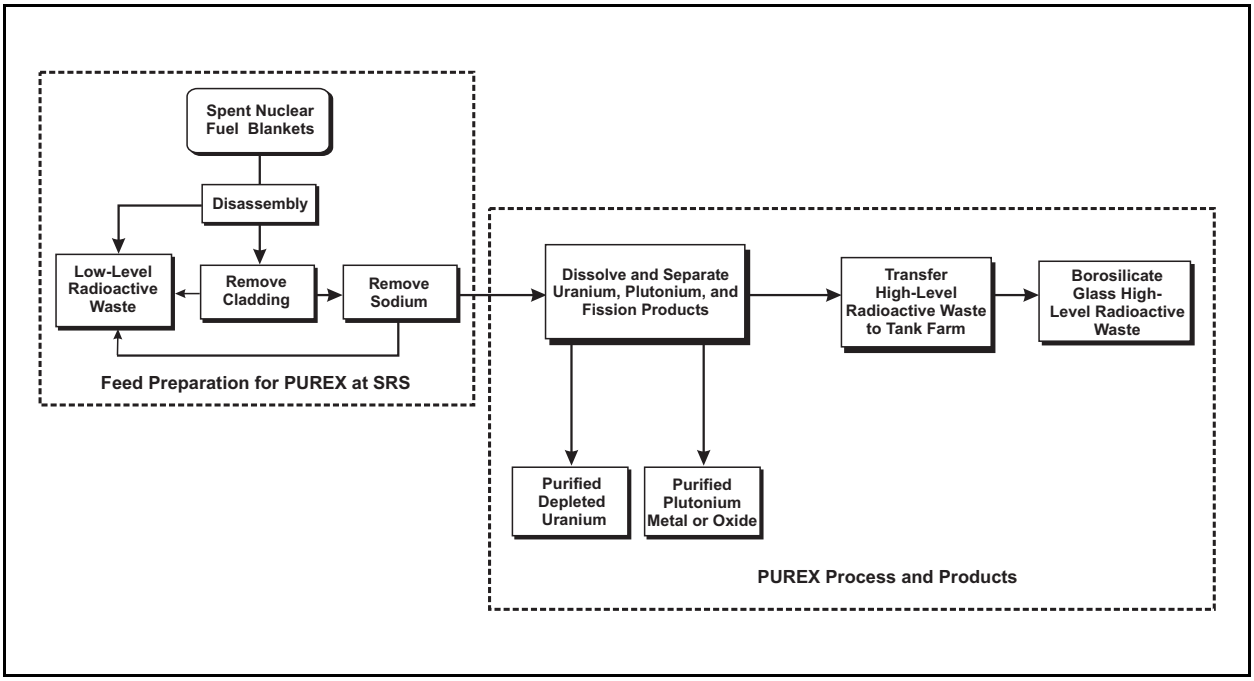

Figure 2–3 PUREX Process Flow Diagram

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.

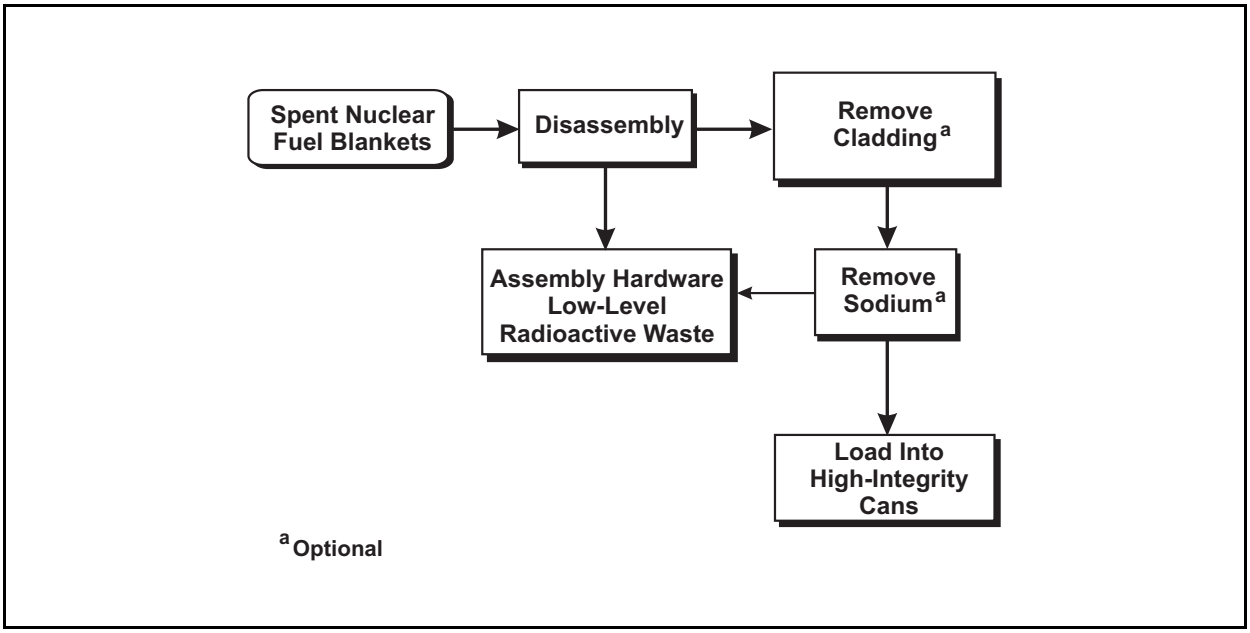

Figure 2–4 High-Integrity Can Process Flow Diagram

2.4.4 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 that would be capable of handling the sodium volatilized from processing chopped driver spent nuclear fuel elements with the cladding intact.

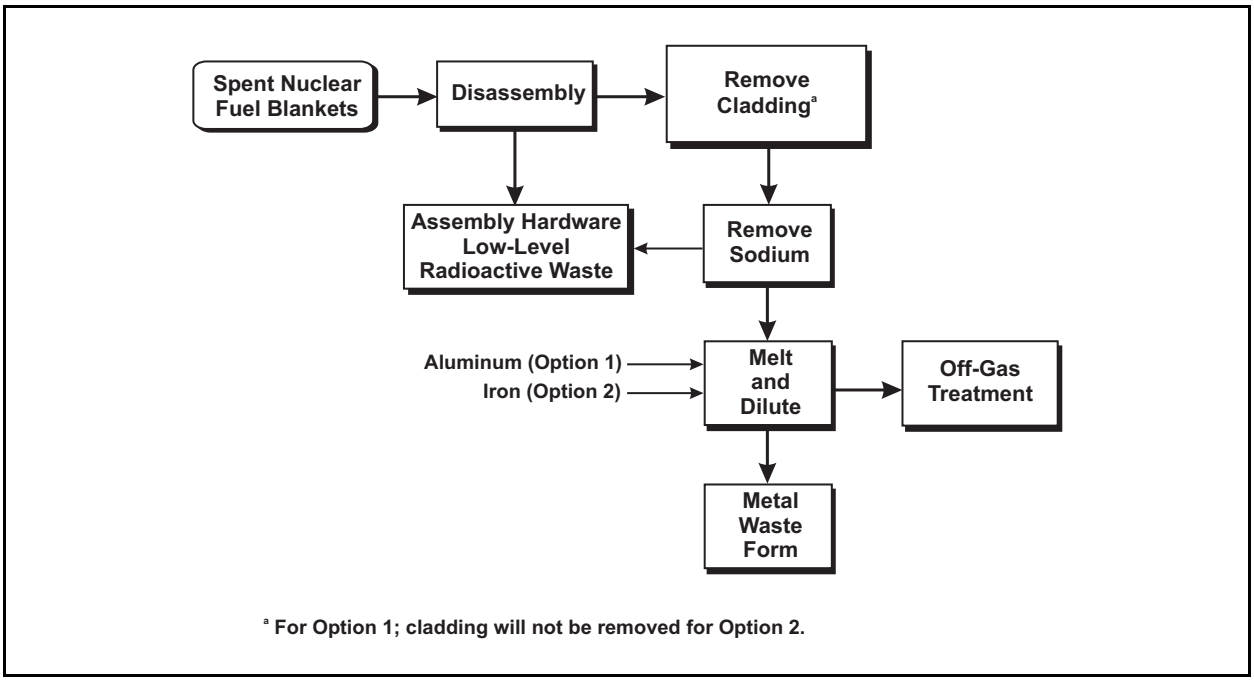

Figure 2–5 Melt and Dilute Process Flow Diagram (Option 1 or 2)

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 some validation of 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 (1,832 °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.

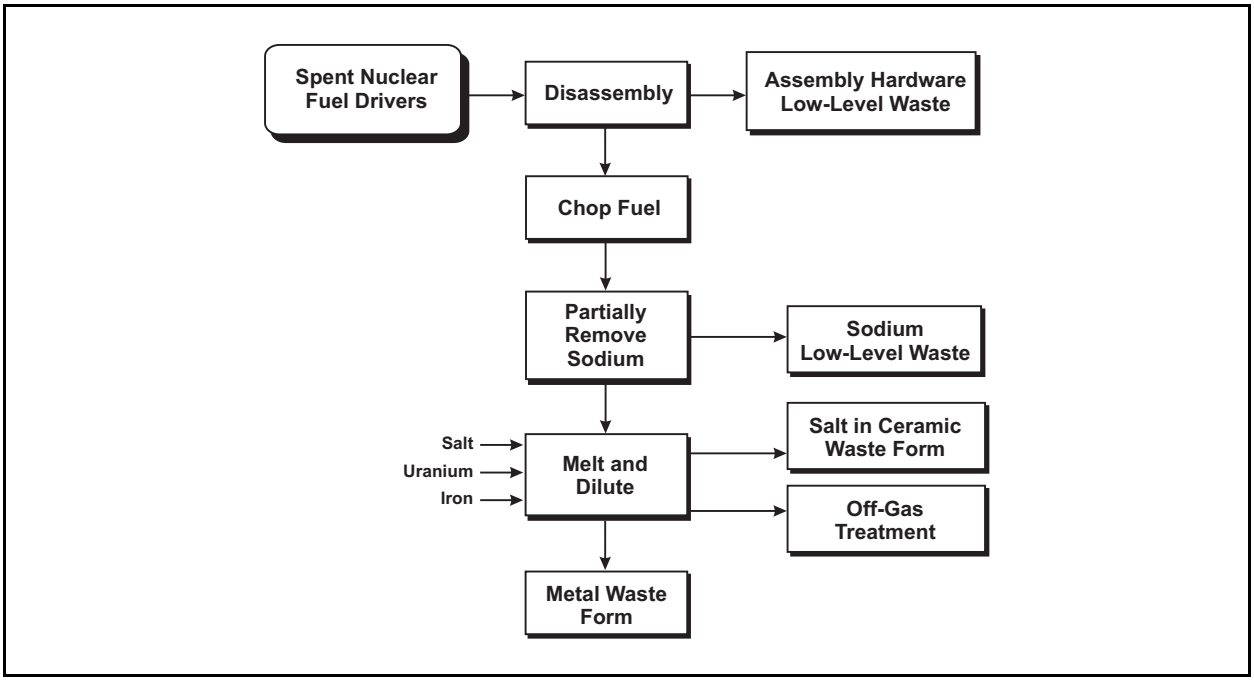

Figure 2–6 Melt and Dilute Process Flow Diagram (Option 3)

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 (2,552 °F) to melt the uranium, after which the steel would slowly be dissolved into the uranium pool. The mixture would be electromagnetically stirred 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.

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. A small portion of 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 volatization of the salt, which would be condensed and partially reused. The salt, which includes sodium in a nonreactive form, would be stabilized in the 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 nitrite fuels cannot be treated using the melt and dilute process because of their high melting points.

2.4.5 Glass Material Oxidation and Dissolution System (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 (1,472 to 1,832 °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.

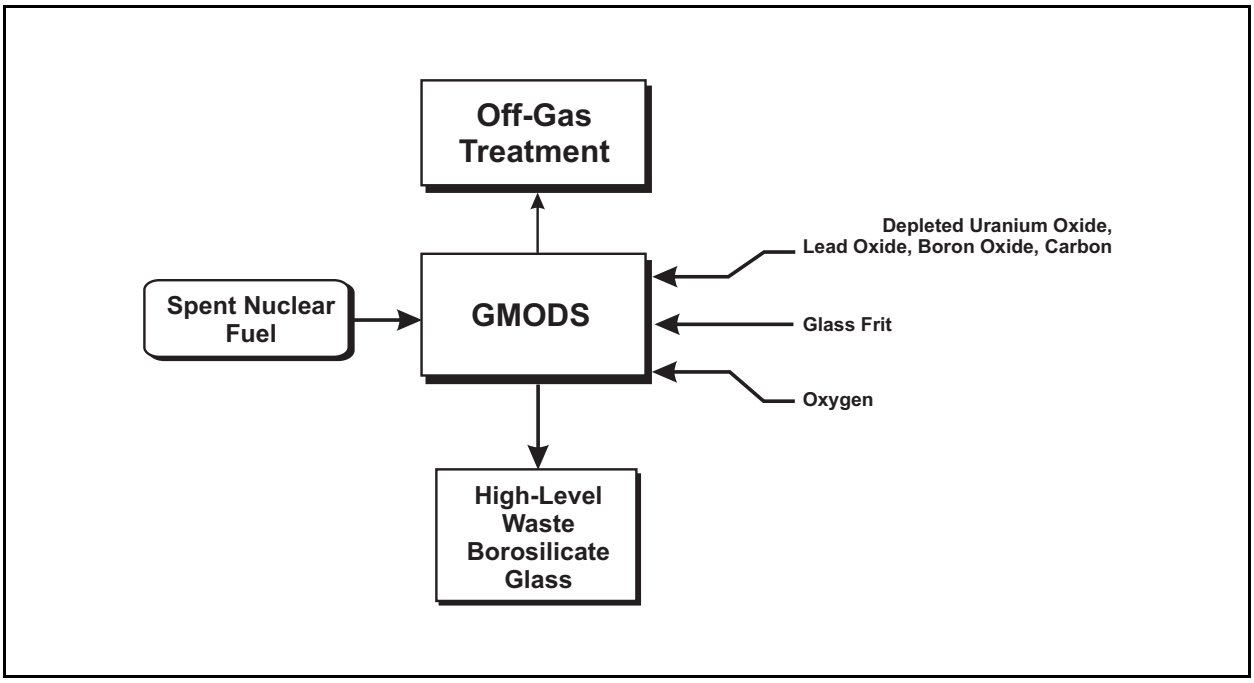

Figure 2–7 Glass Material Oxidation and Dissolution System (GMODS) Process

2.4.6 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 1,600 °C (2,912 °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. **Figure 2–8** illustrates the direct plasma arc-vitreous ceramic treatment process. A description of this process is presented in Appendix C.

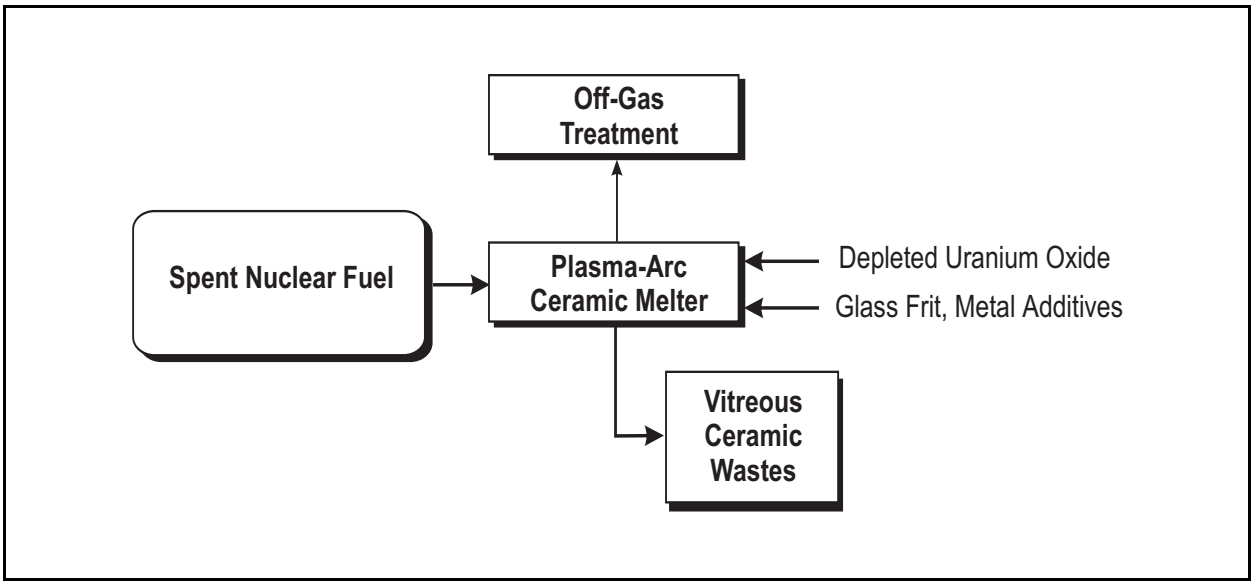

Figure 2–8 Direct Plasma Arc-Vitreous Ceramic Treatment Process Flow Diagram

Metallic fuels such as 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.

2.4.7 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 (2,732 °F) and conversion of metallic fuel and cladding to gaseous chloride compounds; (2) removal of the transuranic chlorides and most of the fission products in a molten zinc chloride bed at approximately 400 °C (752 °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.

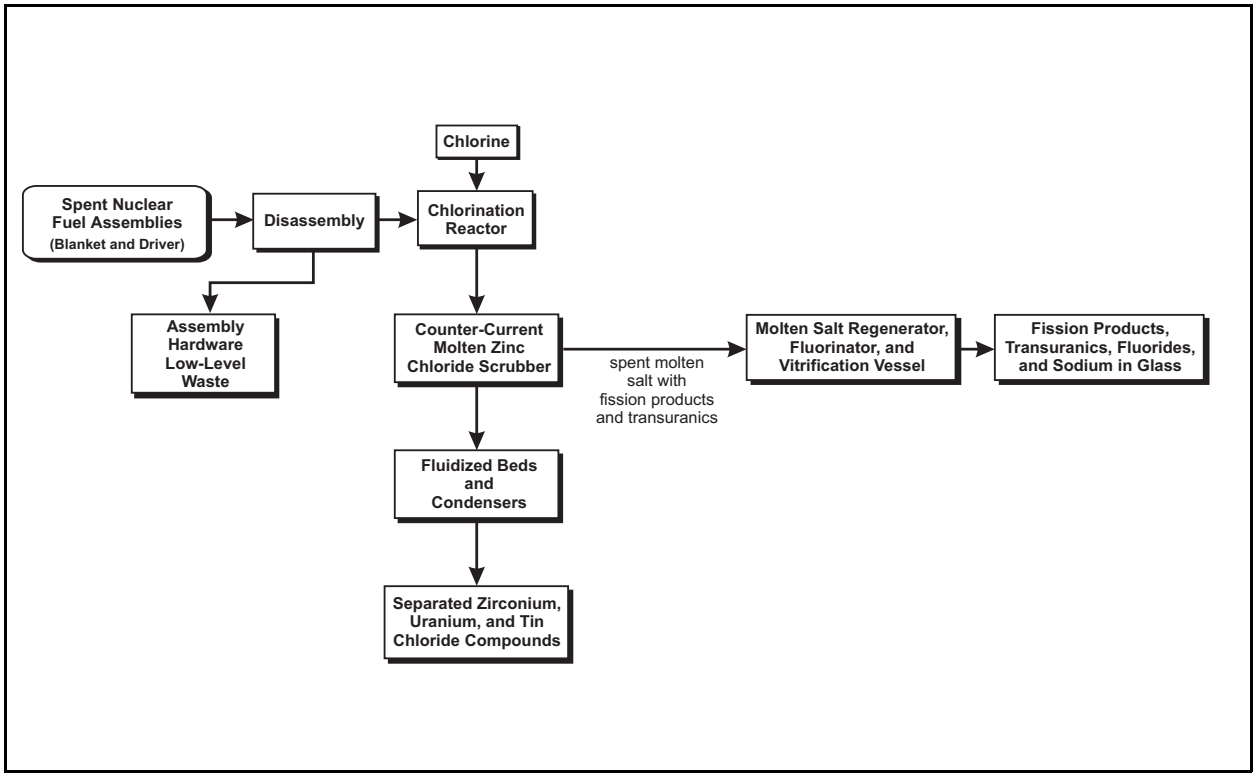

Figure 2–9 Chloride Volatility Process Flow Diagram

2.4.8 Direct Disposal

Direct disposal of sodium-bonded spent nuclear fuel is disposal with minimal preparation (cleaning and conditioning) and no 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 pre-closure operations. At the present time, Resource Conservation and Recovery Act (RCRA) mixed waste (which contains both hazardous and radioactive waste) does not meet the requirements of acceptable waste as identified in the DOE Civilian Radioactive Waste Management Office's March 19, 1999, draft *Waste Acceptance System Requirements Document* (DOE 1998a). 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 U.S. Nuclear Regulatory Commission (NRC) prohibits the disposal of materials that contain or generate explosive, pyrophoric, or chemically reactive substances that could compromise the repository's performance.

2.4.9 Sodium Removal and Disposition

As discussed in 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. Direct disposal also does not call for sodium removal, so the metallic sodium would be a constituent in the disposal package.

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. The sodium removal process is as follows:

The 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 200 °C (392 °F) temperature which causes melting of the sodium, which is drained into a collection tank. After this bulk sodium is removed, the fuel temperature is raised to 500 °C (932 °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. This cesium would be recovered by vacuum distillation of the sodium, taking advantage of the large difference in the boiling points of the 2 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. This machine would mechanically push the fuel pins within the cladding out through the opening created when the cladding ends of the fuel elements were previously cut off. Experience with unirradiated blanket fuel at ANL has shown that the pins could be mechanically pushed out of the stainless steel cladding after all the sodium bond 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. The remaining sodium would be removed during the process as nonreactive salt, stabilized in a ceramic waste form, and disposed of as low-level radioactive waste.

For the less mature technologies, i.e., the GMODS process, the plasma arc process, and the Chloride Volatility process it is expected that sodium removal and disposition would be similar to that for the melt and dilute process.

Table 2–3 provides a summary for sodium removal and sodium 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 ceramic radioactive waste product of the process.
High-Integrity Cans Blanket Fuel	No	Removal	Converted into nonreactive form, separate from the process, and disposed of as low-level radioactive waste.
PUREX Process Blanket Fuel	Yes	Removal	Converted into nonreactive form, separate 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, separate 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 low-level radioactive waste.
Blanket Fuel	Yes	Removal	Converted into nonreactive form, separate from the process, and disposed of as low-level radioactive waste.
Direct Disposal Driver and Blanket Fuel	No	No	Disposed of in metallic reactive form in high-integrity cans.

2.5 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.

2.5.1 Argonne National Laboratory-West

The ANL-W site is a center of nuclear technology development and testing. **Figure 2–10** shows the ANL-W facilities that would be used to treat and manage sodium-bonded spent nuclear fuel. 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 Argonne National Laboratory-West

2.5.1.1 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.

Figure 2–11 Fuel Conditioning Facility at ANL-W

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.

2.5.1.2 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.

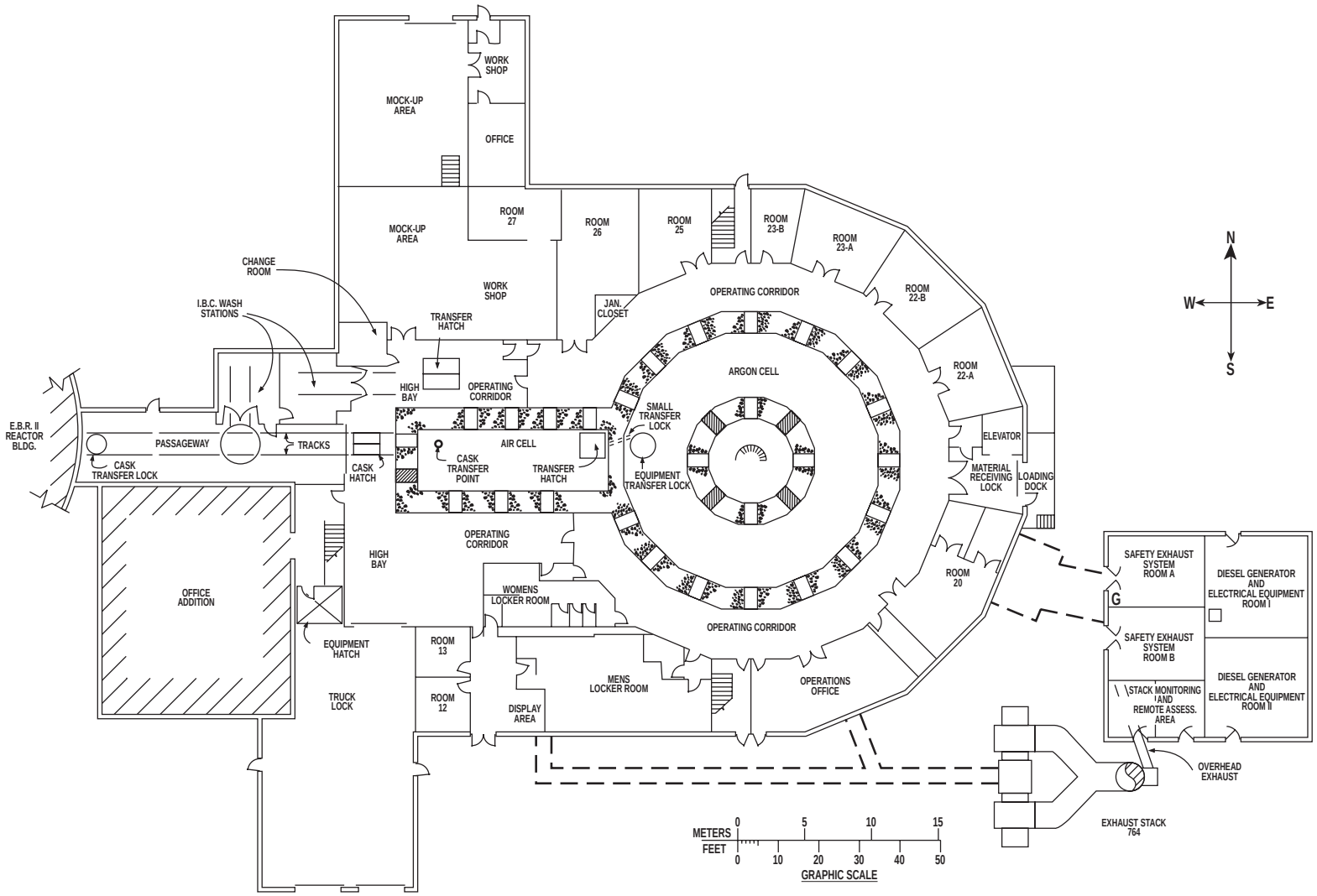

Figure 2–12 Main Floor Layout of the Fuel Conditioning Facility at ANL-W

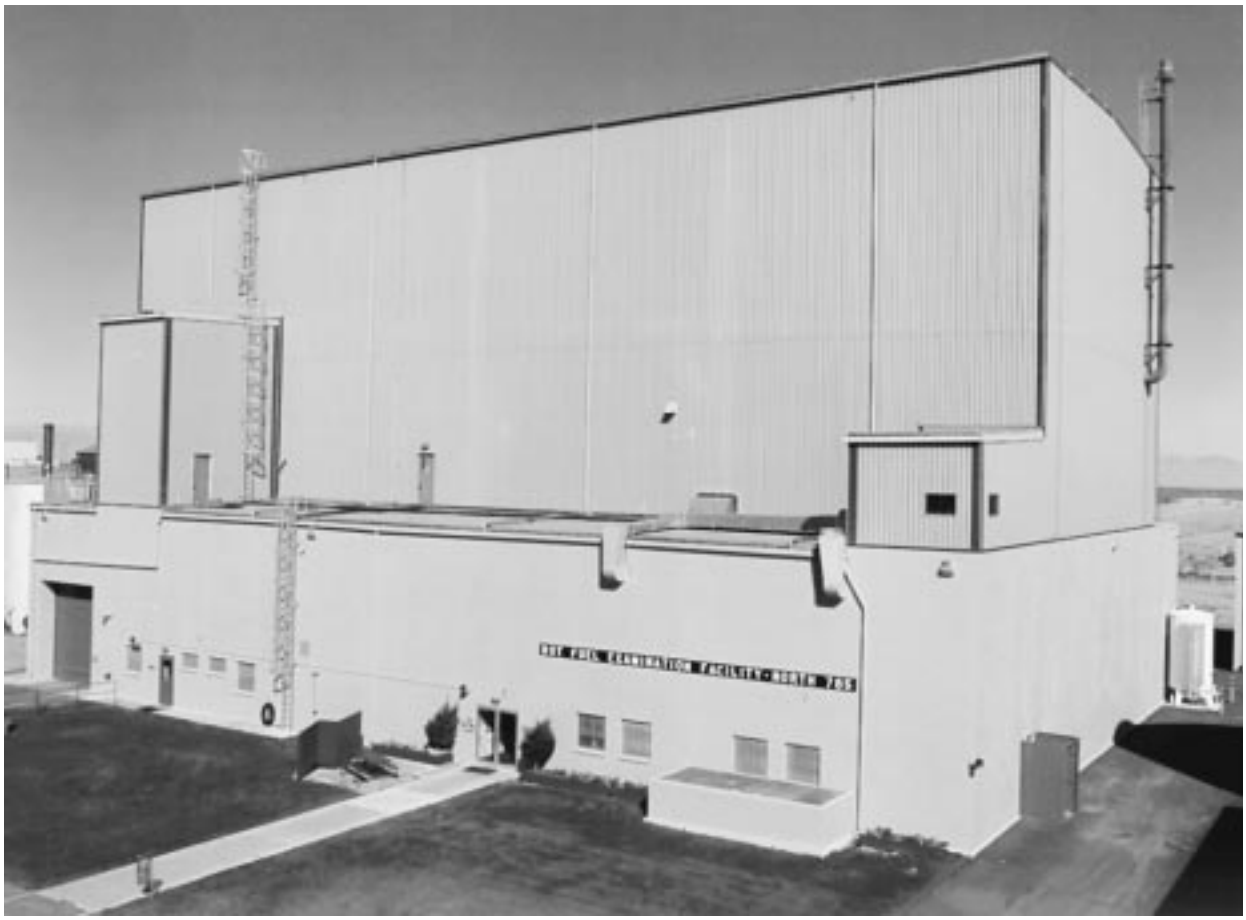

Figure 2–13 Hot Fuel Examination Facility at ANL-W

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.

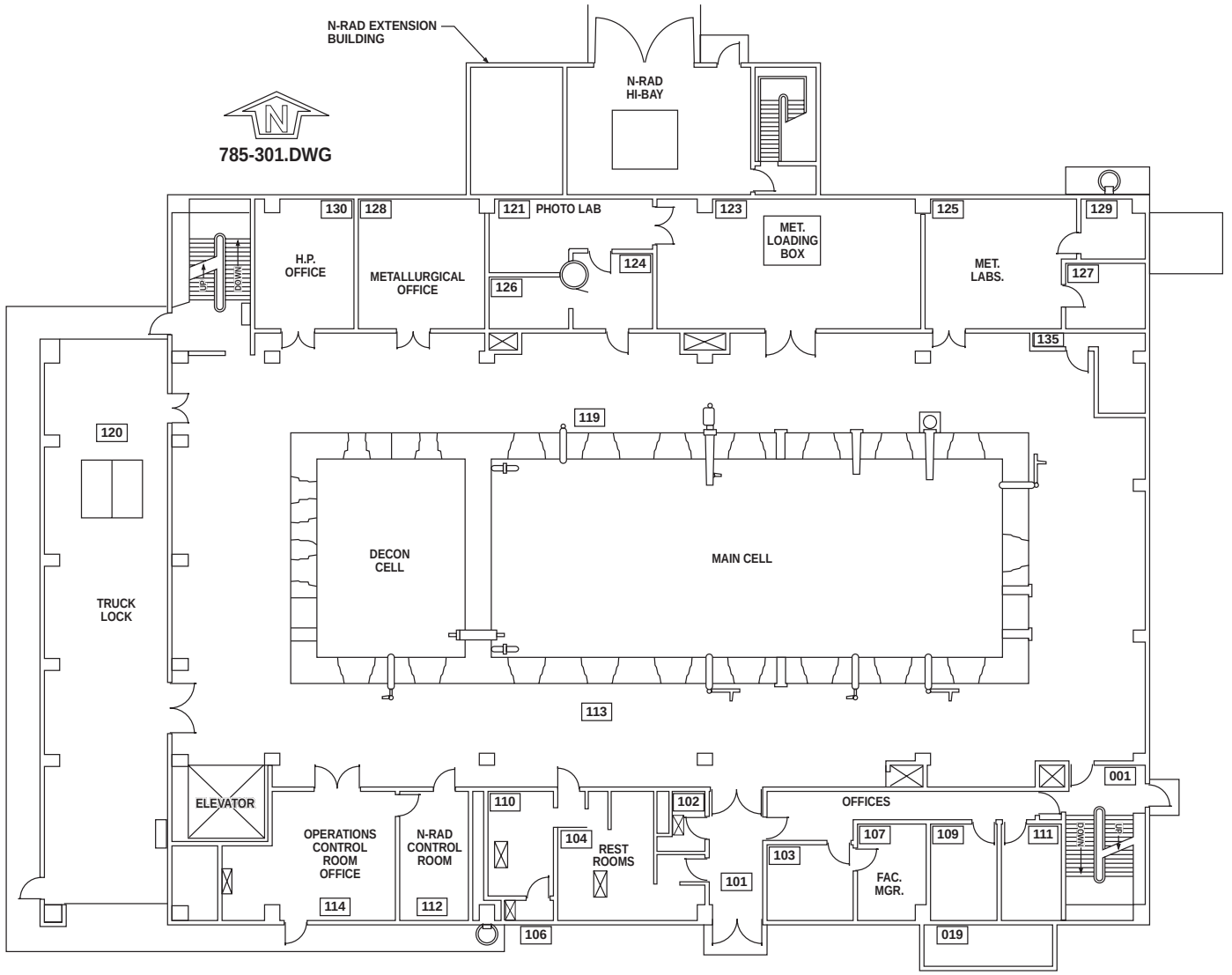

Figure 2–14 Main Floor Layout of the Hot Fuel Examination Facility at ANL-W

2.5.1.3 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

2.5.1.4 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 is currently stored and the facility where high-level radioactive waste from the treatment of the fuel could be stored pending ultimate disposal. Located underground and 0.8 kilometers (0.5 miles) northeast of ANL-W, 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 the RCRA. A major upgrade of the Radioactive Scrap and Waste Facility provides active electrical 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

2.5.2 Idaho Nuclear Technology and Engineering Center at INEEL

The Idaho Nuclear Technology and Engineering Center (formerly known as the Idaho Chemical Processing Plant) is located northeast of the Central Facilities Area at INEEL. It is one of the sites where sodium-bonded spent nuclear fuel is currently 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 1,100 people currently work at INTEC. The facility would be used to package the currently stored sodium-bonded spent nuclear fuel for direct disposal if treatment is not necessary. 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 Idaho Nuclear Technology and Engineering Center at INEEL

The primary facilities at INTEC include:

- The Fluorinel Dissolution Process and Fuel Storage 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 houses both underwater pools and dry storage facilities for spent nuclear fuel storage. 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 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 evaporates, 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 is currently used to convert sodium-bearing waste to granular solid with a smaller volume reduction from liquid to solid.

- The Remote Analytical 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 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 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 wastes in the Draft Idaho High-Level Wastes and Facilities Disposition EIS, which is scheduled to be issued for public comment in September 1999. 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.

2.5.3 Savannah River Site

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 chemically processed 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 Draft Environmental Impact Statement* (DOE 1998) 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 byproducts such as high-level radioactive waste. The high-level radioactive waste, about conversion (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.

2.5.3.1 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 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 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.

2.5.3.1.1 FB-Line

The FB-Line, located on top of the F-Canyon, historically converted plutonium nitrate solution produced in F-Canyon to plutonium-239 metal buttons. Solutions from the F-Canyon are concentrated and purified in the FB-Line. The plutonium is then precipitated, filtered, dried, and finally reduced to a metal 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.

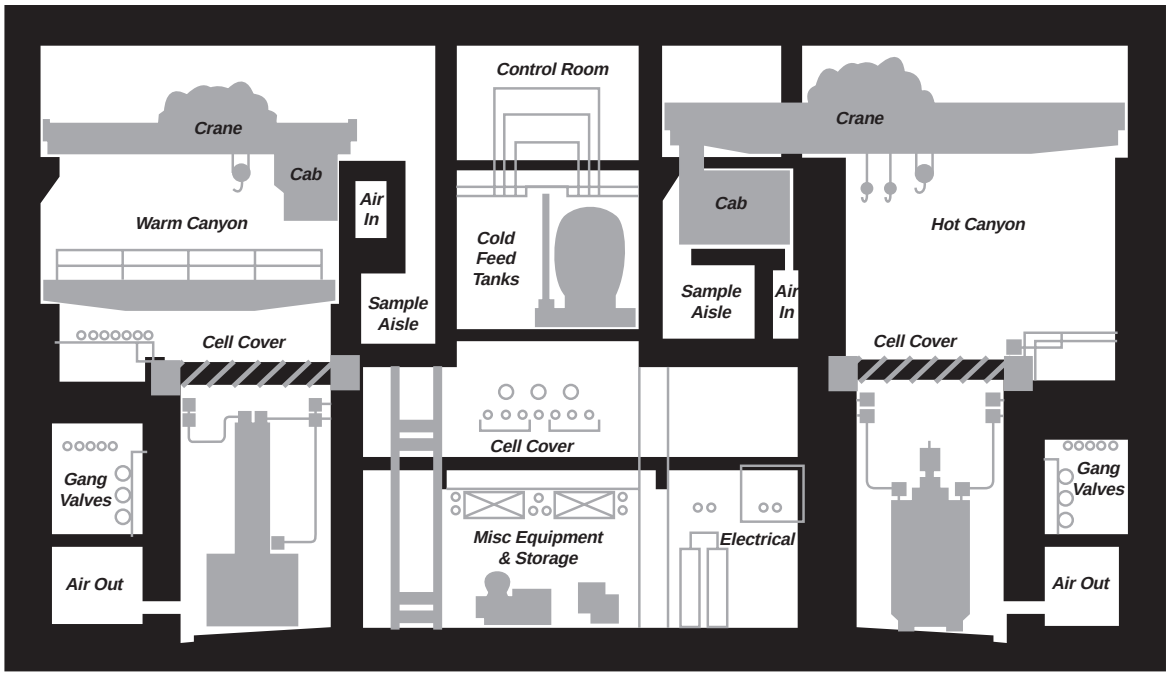

Figure 2–19 F-Canyon Building Sections (Hot Canyon and Warm Canyon) at SRS

DOE also has determined that the FB-Line should be used to stabilize plutonium.

2.5.3.2 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 shutdown 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 1998). 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.

Figure 2–20 Building 105-L at SRS

In the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement*, (DOE 1998), 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.

DOE expects the melt and dilute option would be relatively simple to implement in Building 105-L. The major technical issue for 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 *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement*, (DOE 1998).

2.5.3.3 Defense Waste Processing Facility

The Defense Waste Processing Facility 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 wastes are currently 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 will be stored at SRS until a Federal repository is established.

2.6 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 separately in identifying a preferred alternative and any subsequent Record of Decision. In other words, DOE will consider all combinations of technologies, options, and fuel types, including combinations not among the specific combinations explicitly considered 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.

2.6.1 No Action Alternative

Under the No Action Alternative, 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 are analyzed: (1) the sodium-bonded spent nuclear fuel would continue to be stored indefinitely at its current location in accordance with the Record of Decision (60 FR 28680) for the Programmatic Spent Nuclear Fuel EIS (DOE 1995) and other existing site-specific NEPA documentation or until a technology currently dismissed as an unreasonable alternative because of immaturity (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, e.g., the fuel would be packaged in high-integrity cans with minimal preparation (cleaning and conditioning) and without sodium removal.

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 fuels and they require minimal preconditioning of the sodium-bonded 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.

2.6.2 Alternative 1: Electrometallurgical Treatment of Blanket and Driver Fuel at ANL-W

Under this alternative, all sodium-bonded spent nuclear fuel (both driver and blanket spent nuclear fuels), 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 subassembly 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.

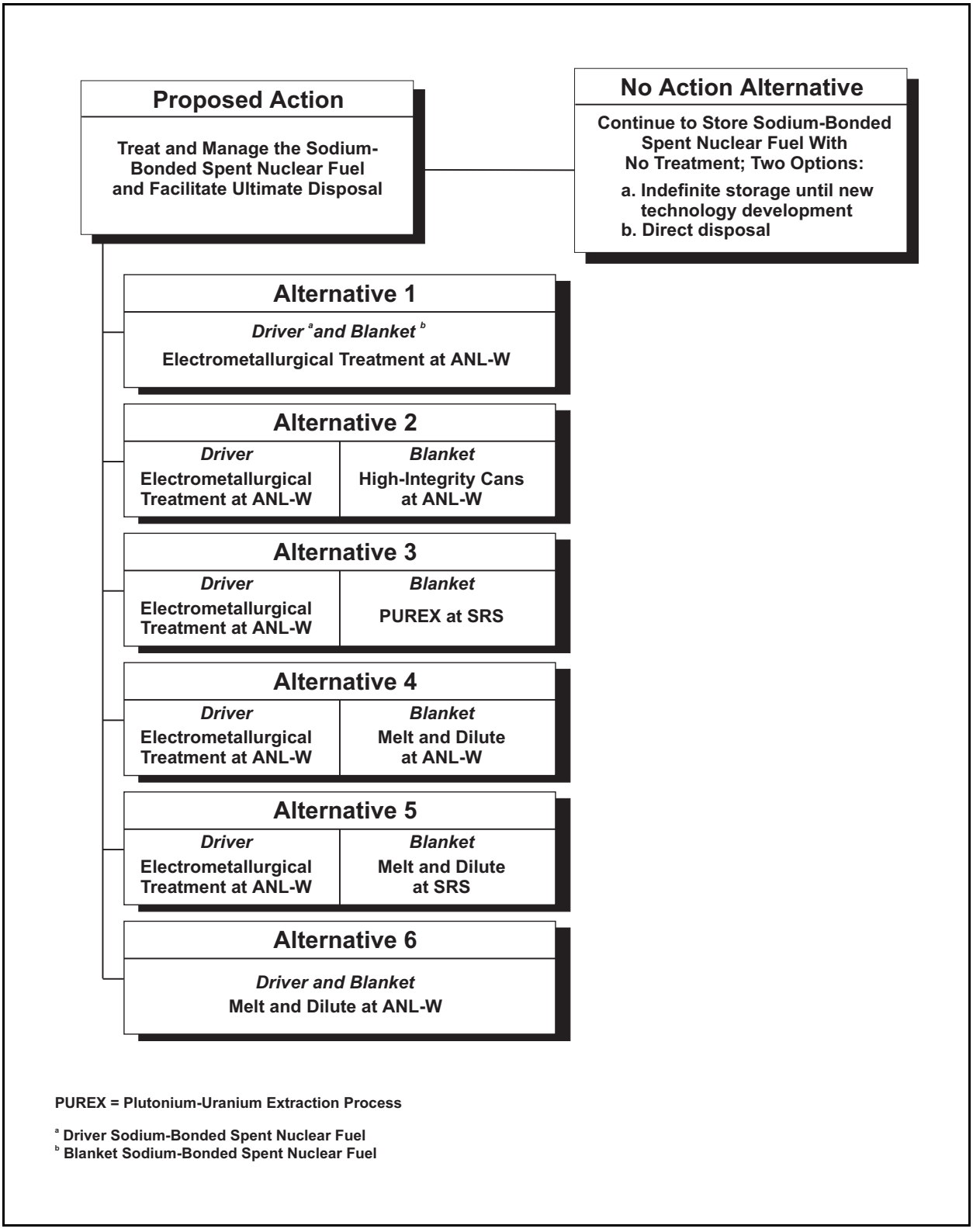

Figure 2–22 Proposed Action and Alternatives

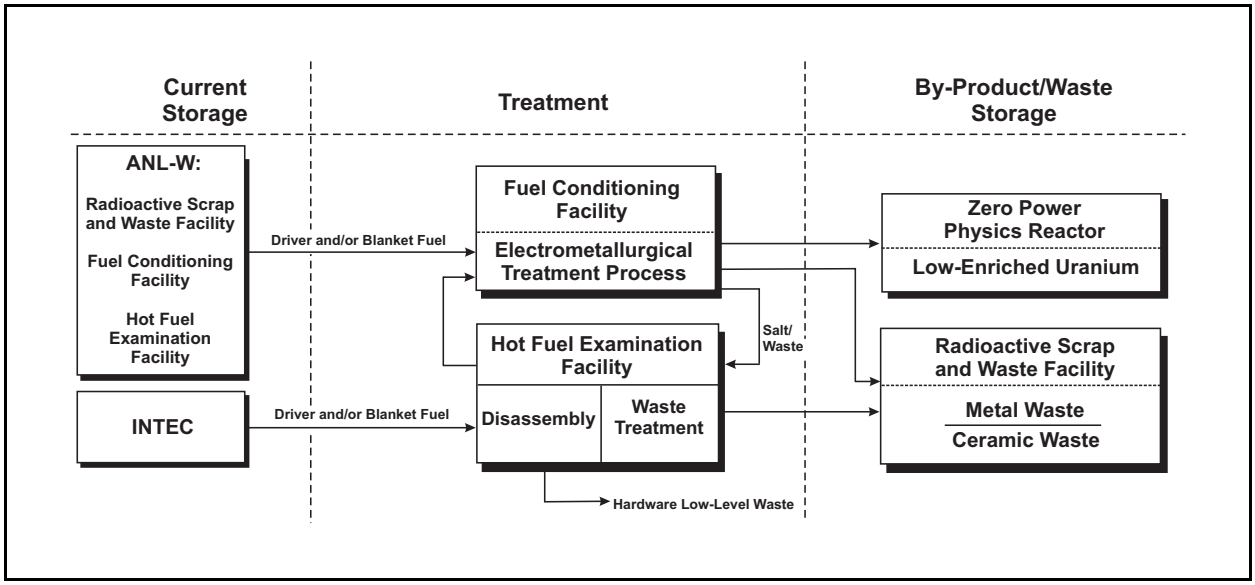

Figure 2–23 Schematic for Driver and Blanket Spent Nuclear Fuel Treatment in Alternative 1

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 metallic 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.

2.6.3 Alternative 2: Remove Sodium and Package Blanket Fuel in High-Integrity Cans and Treat (Electrometallurgical) 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.

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. **Figure 2–24** illustrates the steps of the process for the blanket sodium-bonded spent nuclear fuel.

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.

Implementing this alternative at the Hot Fuel Examination Facility would require the installation of equipment for sodium removal activities. No new equipment would be needed for the electrometallurgical treatment of the driver sodium-bonded spent nuclear fuel under this alternative.

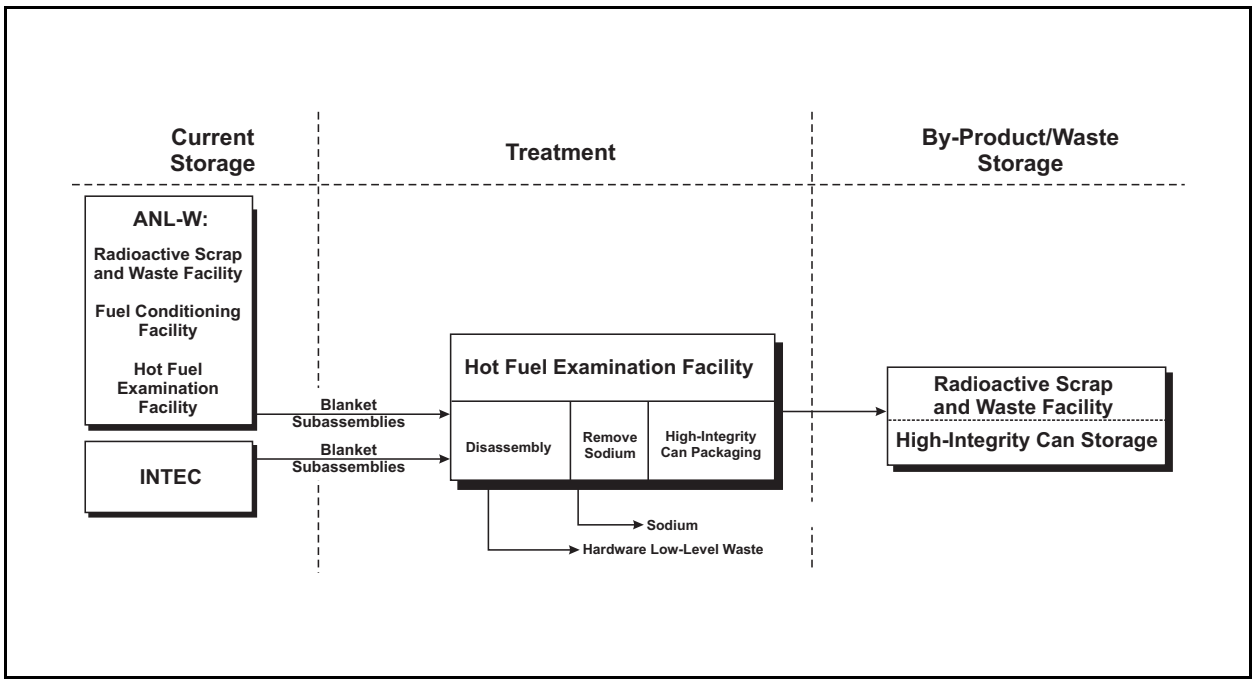

Figure 2–24 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 2

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.

2.6.4 Alternative 3: Declad and Clean Blanket Fuel and Treat (Electrometallurgical) 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 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.

Figure 2–25 illustrates the process steps for the blanket spent nuclear fuel at the Hot Fuel Examination Facility.

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.

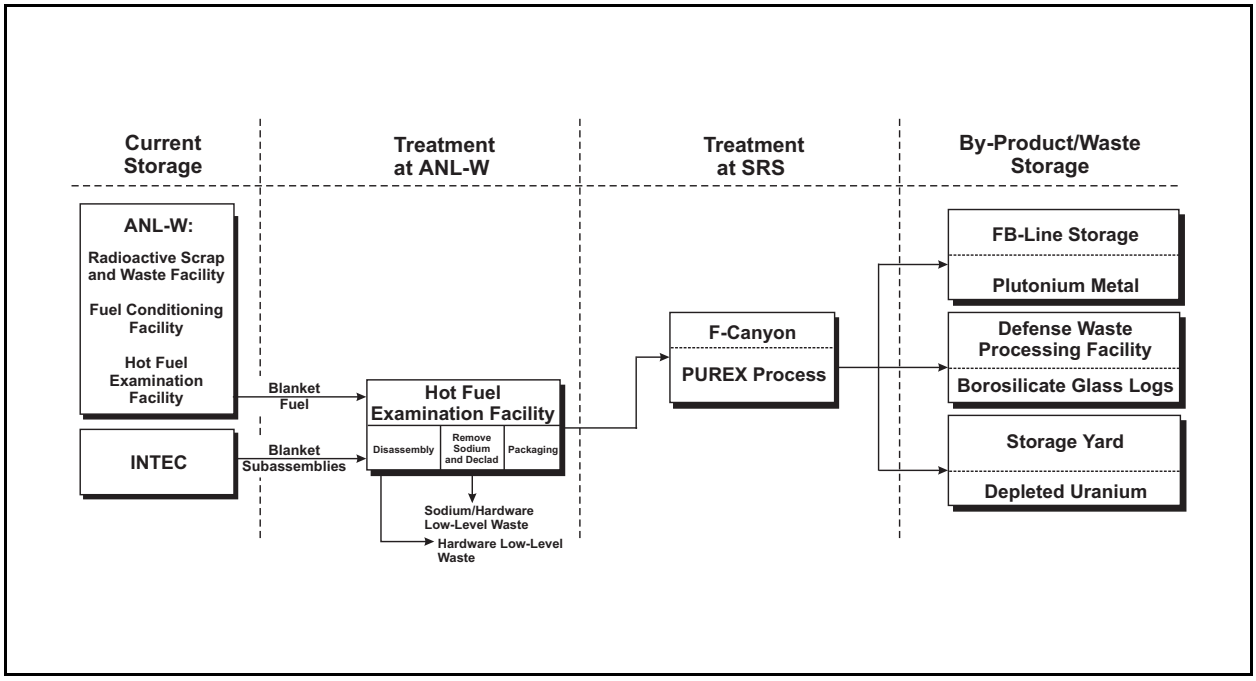

Figure 2–25 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 3

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. Separated plutonium in metal form would be stored at an SRS vault. Depleted uranium would be transferred to a storage yard for depleted uranium at the site.

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. 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.

2.6.5 Alternative 4: Melt and Dilute Blanket Fuel and Treat (Electrometallurgical) 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 the facility 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.

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. **Figure 2–26** illustrates the process steps for the sodium-bonded blanket spent nuclear fuel.

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.

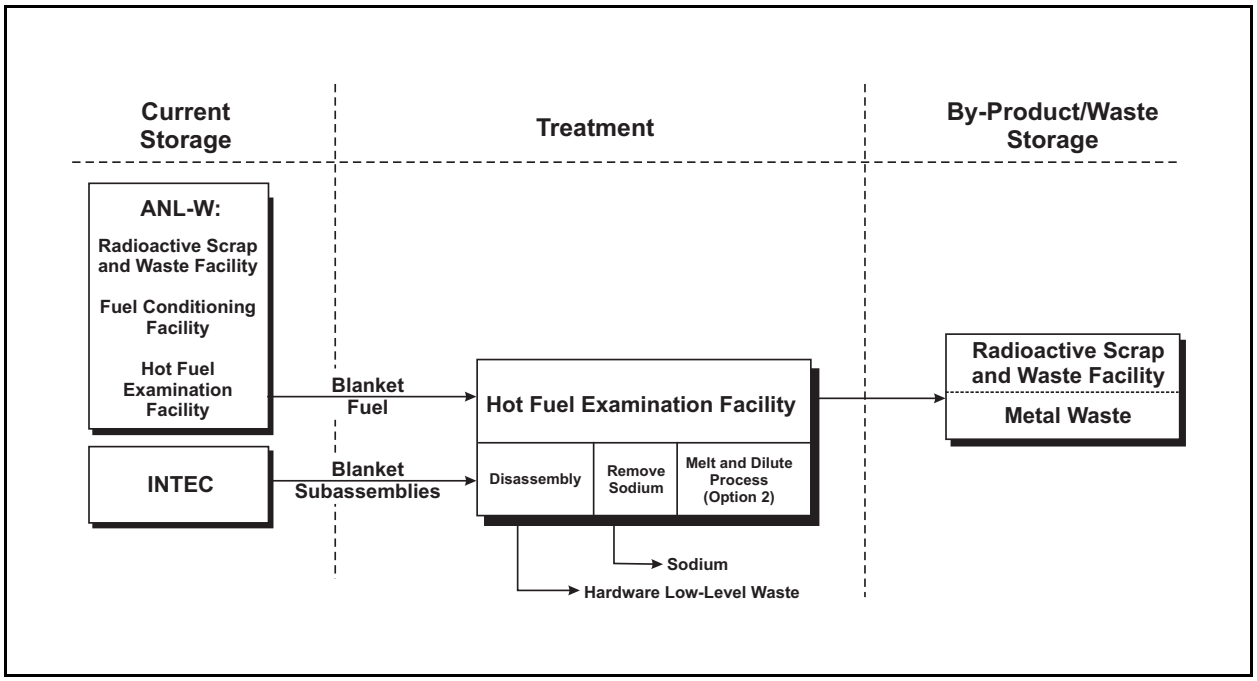

Figure 2–26 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 4

Metal 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.

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 seven years. Treatment of the driver spent nuclear fuel could start as early as 2000 and could be completed in approximately seven years.

2.6.6 Alternative 5: Declad and Clean Blanket Fuel and Treat (Electrometallurgical) 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 the 105-L Building at SRS and treated using the melt and dilute Option 1 process, as described in Section 2.3.4.

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.

Figure 2–27 illustrates the process steps for the blanket spent nuclear fuel.

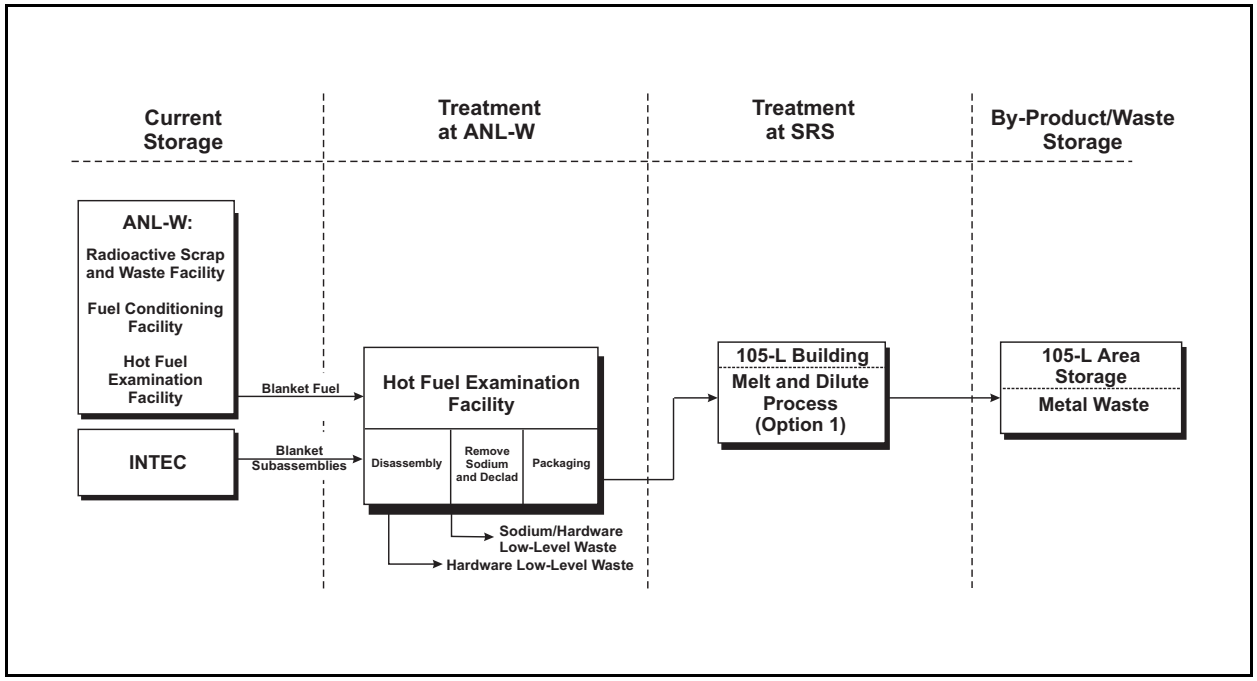

Figure 2–27 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 5

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 the 105-L Building 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 *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement* (DOE 1998b).

Metal waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be stored at the 105-L Area storage pending ultimate disposal.

Treatment of the driver spent nuclear fuel at ANL-W could start in 2000 and could be completed in approximately seven years. 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.

2.6.7 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 of oxide, carbide, and nitride fuel, which would not be treated under the alternative. **Figure 2–28** illustrates the steps for the alternative.

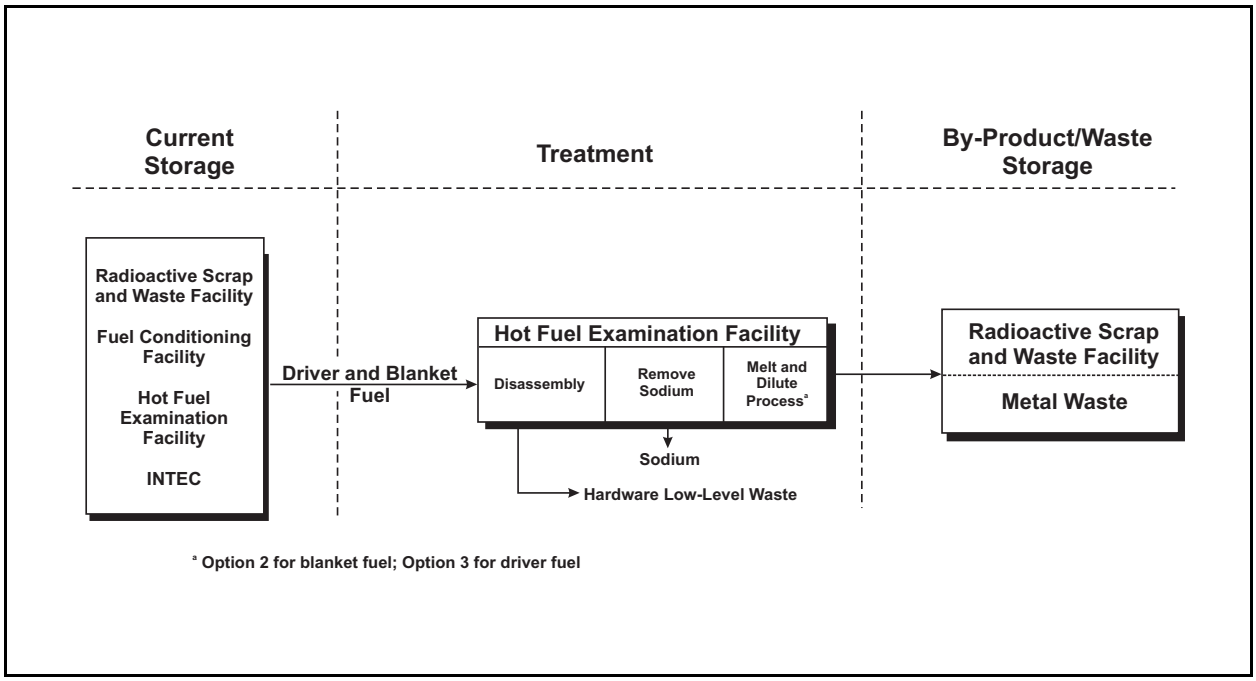

Figure 2–28 Schematic for Driver and Blanket Spent Nuclear Fuel in Alternative 6

Removal of the sodium from the blanket 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 metal 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.

The melt and dilute process at ANL-W could start as early as 2005 and would take approximately 10 years to be completed for all (driver and blanket) fuels.

2.7 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 aluminum-clad spent nuclear fuel at SRS (DOE 1998b). 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 fuels. 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.

Glass Material Oxidation and Dissolution System Process

The GMODS process, although similar to the melt and dilute process because of its thermal treatment, has not been developed beyond laboratory scale. Several developmental steps would be required before it can be deemed a mature process. These include: detailed process development, resolution of containment concerns, testing, and pilot plant demonstration to address technology risks (for example, 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 would be required to treat the radioactive elements volatilized at 1,000 °C (1,832 °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 plasma arc-vitreous ceramic process is being used for the vitrification of mixed wastes. 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 the 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 Cladded 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 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.

2.8 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 suitably packaged for disposal. DOE has drafted preliminary acceptance criteria which are being used to assess the feasibility of DOE spent nuclear fuel disposition options (DOE 1998a). If the repository is developed, final acceptance criteria will not be available until after 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.

In order to ensure that the treatment option DOE could select will produce 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 its most 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.

2.9 PREFERRED ALTERNATIVE

When the Notice of Intent to prepare this EIS was published in the *Federal Register* on February 22, 1999 (64 FR 8553), the proposed action was the electrometallurgical treatment of DOE's inventory of sodium-bonded spent nuclear fuel in the Fuel Conditioning Facility at ANL-W. In response to public comments received during the scoping process, DOE has reformulated the scope of the EIS to address more generally the treatment and management of DOE sodium-bonded spent nuclear fuel and to separate the technical analysis of blanket and driver spent nuclear fuel. Under the revised proposed action, several technology alternatives, including various combinations for driver and blanket spent nuclear fuel, have been analyzed in this Draft EIS. Information developed in the course of preparing this EIS suggests that alternative technologies may have certain advantages (e.g., cost) for some or all fuel. Accordingly, DOE has no preferred alternative at this time. DOE will consider the environmental analyses in this EIS, public comments, and the findings of the independent cost study and the nonproliferation report, as well as other program policy factors, in determining a preferred alternative in the Final EIS.

2.10 SUMMARY COMPARISON OF ENVIRONMENTAL IMPACTS OF THE REASONABLE ALTERNATIVES

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 the 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, DOE has determined that the proposed action would have minimal or no impacts on the remaining environmental areas (e.g., land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources) at the proposed sites. This is because the proposed facilities already exist so, except for internal building modifications and new equipment installation, no construction activities would be required.

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**.

2.10.1 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 safe, secure, indefinite storage at current locations, 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 with minimal preparation.

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 indefinite 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. The impacts summarized below would be applicable to both options considered under the No Action Alternative.

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. 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 623 indirect jobs. 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 only 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 annual dose to the population within 80 kilometers (50 miles) from these releases would be 0.022 person-rem per year. The risk that an average individual would develop a fatal cancer from this

exposure would be 0.000011, or a probability of one cancer fatality in 90,900 years. 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 0.00077 millirem per year, and the risk of developing a fatal cancer from this exposure would be 3.9×10^{-10} (once in 2.5 billion years). The average worker would receive 60 millirem per year, and the risk of developing a cancer from this exposure would be 0.000024, or once in 41,666 years.

The annual cancer risk from postulated accident conditions under either option of the No Action Alternative at ANL-W would be 6.9×10^{-8} for the population within 80 kilometers (50 miles). Since INTEC is further away from the INEEL site boundary and major population centers compared to ANL-W, the accident impacts would be less than those presented for ANL-W. The annual cancer risk for the maximally exposed offsite individual would be 6×10^{-10}, and for the noninvolved worker it would be 1.9×10^{-10}.

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 values, indicate that under either option of the No Action Alternative the worst postulated accident conditions would result in less than Emergency Response Planning Guideline-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 and INTEC. These include low-level radioactive, transuranic, mixed, hazardous, and nonhazardous wastes. They are associated with the operation of the facilities where the sodium-bonded spent nuclear fuel is stored. High-level radioactive waste in metal and ceramic forms generated as a result of completing waste processing of the electrometallurgical treatment demonstration project 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 wastes are presented in Table 2-3.

Transportation

No offsite transportation activities would occur under either option of the No Action Alternative.

2.10.2 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 (Alternative2); 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 have very small impacts on air quality, water resources, socioeconomics, public and occupational health and safety, 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 effluents, as well as the associated exposures to workers and the public, are 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 consideration in this EIS, the volume of high-level radioactive waste for all the considered alternatives is an important consideration. All the proposed action alternatives result in a decrease in high-level radioactive waste volume as compared to the direct disposal No Action Alternative: 45 percent (Alternative 1); 70 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 will not change as a result of the proposed action.

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 per year of elemental tritium and 11,600 (Alternative 1) to 32,250 (Alternative 6) curies per year of krypton-85.

Radiological gaseous emissions at SRS would be 54 (Alternative 5) to 162 (Alternative 3) curies per year of elemental tritium and 399 (Alternative 5) to 1,188 (Alternative 3) curies per year 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.

No changes are 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 2-3 indicates some small quantities of tritium, strontium, ruthenium, and isotopes of uranium and plutonium. 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 2-3, the risk is small for all alternatives considered under the proposed action.

Radiological Exposures

Comparing alternatives at ANL-W, the annual population dose from routine gaseous radioactive releases would range from 0.0029 person-rem (Alternative 1) to 0.012 person-rem (Alternative 6), with a latent cancer fatality risk in the range of 1.5×10^{-6} to 6.0×10^{-6}, respectively.

The 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 annual dose to an average individual within the 80-kilometer (50-mile) population would range from 0.000012 millirem (Alternative 1) to 0.00051 millirem (Alternative 6), with a latent cancer fatality risk in the range of 6.0×10^{-12} to 2.6×10^{-11}, 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.

Comparing alternatives at SRS, the maximum population dose from routine gaseous radioactive releases would range from 0.0076 person-rem per year (Alternative 5) to 0.02 person-rem for the whole treatment period (Alternative 3), corresponding to additional latent cancer fatalities in the range of 3.8×10^{-6} to 0.000010, respectively.

The maximum dose to the maximally exposed offsite individual would range from 0.00010 millirem per year (Alternative 5) to 0.00051 millirem (Alternative 3) for the whole treatment period, with a latent cancer fatality risk in the range of 5.0×10^{-11} to 2.6×10^{-10}, respectively. The dose to an average individual within the 80-kilometer (50-mile) population would range from 0.000011 millirem per year (Alternative 5) to 0.000024 millirem (Alternative 3), with a latent cancer fatality risk in the range of 5.3×10^{-12} to 1.2×10^{-11}, respectively.

The maximum collective dose to workers at SRS would be 50 person-rem per year (Alternative 5). This corresponds to additional latent cancer fatalities of 0.075. 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 0.00010 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 0.0088×10^{-3} (Alternative 6, driver fuel, beyond-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 (Alternative 1, 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} (Alternative 1, driver fuel, beyond-design-basis

earthquake). The highest annual latent cancer fatality risk for the noninvolved worker would be 1.7×10^{-9} (Alternative 1, 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 0.013 (Alternative 5, blanket fuel, loss of cooling water). 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 3.1×10^{-6} (Alternative 5, blanket fuel, loss of power).

Hazardous Chemical Exposures

Hazardous chemical impacts from normal operation 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, if any.

Hazardous chemical impacts under accident conditions, evaluated in terms of comparison to Emergency Response Planning Guideline values, indicate that under the proposed action, all postulated hazardous chemical releases would not result in worse than Emergency Response Planning Guideline-2 conditions for a worker, and worse than Emergency Response Planning Guideline-1 conditions for a maximally exposed offsite individual at either ANL-W or SRS.

Waste Management

Table 2-3 presents a comparison of the volumes of high-level radioactive, low-level radioactive, and transuranic wastes generated by each of the alternatives. The alternatives would generate from 37 to 84 percent less high-level radioactive waste as compared to the No Action Alternative option of the direct disposal of spent nuclear fuel. Alternative 2 would also generates less low-level radioactive waste when compared to the No Action Alternative. In comparison, all other alternatives would generate greater volumes of low-level radioactive waste. Each of the alternatives would generate more transuranic waste, but Alternatives 1, 2, 4, and 6 would only exceed this waste volume by a range of 7 to 41 percent. Alternatives 3 and 5 would generate significantly greater volumes of transuranic waste, between 2.3 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 shown to meet current criteria. It is expected, however, that other waste forms (e.g., ceramic, metal, and high-integrity cans not containing metallic sodium) would also be suitable for repository disposal.

Transportation

The transportation activities under Alternatives 1, 2, 4, and 6 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 4.7×10^{-5} person-rem; the dose to the public would be 3.5×10^{-4} person-rem. Accordingly, incident-free transportation activities would result in 1.9×10^{-8} latent cancer fatalities among transportation workers and 1.7×10^{-7} latent cancer fatalities in the total affected population over the duration of the transportation activities.

The dose to the population from postulated accidents from these activities would be less than 1×10^{-12} person-rem, resulting in less than 1×10^{-15} latent cancer fatalities. Nonradiological traffic fatalities would be 8.2×10^{-7}.

Transportation activities under Alternatives 3 and 5 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 2×10^{-3} person-rem; the dose to the public would be 0.013 person-rem. Accordingly, incident-free transportation activities would result in 7.9×10^{-7} latent cancer fatalities among transportation workers and 6.1×10^{-6} latent cancer fatalities in the total affected population over the duration of the transportation activities. Nonradiological fatalities among the public from vehicle emissions during intersite transportation would be 1.96×10^{-4}.

The dose to the population from postulated accidents from these activities would be 3.0×10^{-6} person-rem, resulting in 1.5×10^{-9} latent cancer fatalities. Nonradiological traffic fatalities would be 0.002.

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/year)	Negligible impact Tritium: 460[a] Krypton-85: 7,120[a]		Negligible impact Tritium: 770 Krypton-85: 11,600		Negligible impact Tritium: 809 Krypton-85: 11,860	
Water Resources - Radiological liquid effluents	No impact No liquid effluent		No impact No liquid effluent		No impact No liquid effluent	
Socioeconomics	Loss of 350 direct jobs and 623 indirect jobs; no measurable impact		Workforce maintained; No impact		Workforce maintained; No impact	
Public and Occupational Health and Safety						
• Project duration	35 years		13 years		9 years	
• Normal operation	***Person-rem/year***[c]	***LCF***	***Person-rem/year***	***LCF***	***Person-rem/year***	***LCF***
- Population dose	0.022	0.000011	0.0029	1.5×10^{-6}	0.0031	1.6×10^{-6}
- MEI	0.00077	3.9×10^{-10}	0.00034	1.7×10^{-10}	0.00038	1.9×10^{-10}
- Average individual	0.000092	4.6×10^{-11}	0.000012	6.0×10^{-12}	0.000013	6.6×10^{-12}
- Total worker	22	0.0088	22	0.0088	22	0.0088
- Average worker	60	0.000024	60	0.000024	60	0.000024
Hazardous chemicals						
- MEI / worker (impacts)	None					
• Accidents Maximum annual cancer risk (per year)						
- 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]		1.5×10^{-8} (DBA); 1.5×10^{-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	
- Worker	Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
Environmental Justice	No disproportionately high and adverse impact to minority and low-income populations					
Waste Management (cubic meters)						
• High-level radioactive wastes	152 (Direct disposal SNF volume)[d]		84.3		43.9[e]	
• Low-level radioactive wastes	812		861		733.7	
• Transuranic wastes	10		14.1		10.7	
Transportation						
• Incident-free	***Person-rem***	***LCF***	***Person-rem***	***LCF***	***Person/rem***	***LCF***
- Population	No impact		0.00035	1.7×10^{-7}	0.00035	1.7×10^{-7}
- Workers			4.7×10^{-5}	1.9×10^{-8}	4.7×10^{-5}	1.9×10^{-8}
• Accidents						
- Population			1.1×10^{-12}	1.0×10^{-15}	1.1×10^{-12}	1.0×10^{-15}

Alternative 1: *Electrometallurgical Treatment of Blanket and Driver Spent Fuel at ANL-W*
Alternative 2: *Package Blanket Spent Fuel in High-Integrity Cans and Treat Driver Fuel at ANL-W*
Alternative 3: *Declad and Clean Blanket Spent Fuel and Treat Driver Spent Fuel at ANL-W; PUREX Process Blanket Fuel at SRS*
Alternative 4: *Melt and Dilute Blanket Spent Fuel and Treat Driver Spent Fuel at ANL-W*
Alternative 5: *Declad and Clean Blanket Spent Fuel and Treat Driver Spent Fuel at ANL-W; Melt and Dilute Blanket Fuel at SRS*
Alternative 6: *Melt and Dilute Blanket and Driver Spent Fuel at ANL-W*

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: 399		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	
Workforce maintained; no impact		Workforce maintained; no impact		Workforce maintained; no impact		Workforce maintained; no impact		Workforce maintained; no impact		Workforce maintained; no impact	
9 years		Less than 1 year		12 years		9 years		3 years		10 years	
Person-rem/year	***LCF***	***Person-rem/year***	***LCF***	***Person-rem/year***	***LCF***	***Person-rem/year***	***LCF***	***Person-rem/year***	***LCF***	***Person-rem/year***	***LCF***
0.0031	1.6×10^{-6}	0.02	0.000010	0.0031	1.6×10^{-6}	0.0031	1.5×10^{-6}	0.0076	3.8×10^{-6}	0.012	6.0×10^{-6}
0.00038	1.9×10^{-10}	0.00051	2.6×10^{-10}	0.00038	1.9×10^{-10}	0.00038	1.9×10^{-10}	0.00010	5.0×10^{-11}	0.002	1.0×10^{-9}
0.000013	6.6×10^{-12}	0.000024	1.2×10^{-12}	0.000013	6.6×10^{-12}	0.000013	6.6×10^{-12}	0.000011	5.5×10^{-12}	0.000051	2.6×10^{-11}
22	0.0088	38	0.015[a]	22	0.0088	22	0.0088	50	0.02	22	0.0088
60	0.000024	250	0.0001[a]	60	0.000024	60	0.000024	500	0.0002	60	0.000024
None	None	Small	Small	None	None	None	None	Small	Small	None	None
5.6×10^{-6} (DBA); 0.000013 (BDBA)		0.00017 (DBA)		0.00022 (DBA); 0.000013 (BDBA)		5.6 x 10^{-6} (DBA); 0.000013 (BDBA)		0.013		0.0088 (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 x 10^{-8} (DBA); 2.2 x 10^{-7} (BDBA)		6.6 x 10^{-6}		0.000076 (DBA)	
1.5×10^{-8} (DBA); 1.5×10^{-9} (BDBA)		4.8×10^{-7} (DBA)		4.9×10^{-8} (DBA); 1.5×10^{-9} (BDBA)		1.5 x 10^{-8} (DBA); 1.5 x 10^{-9} (BDBA)		3.4 x 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 and low-income populations											
24.3 (18.7 at ANL-W, 5.6 at SRS)				64.3		95.32 (18.7 at ANL-W, 76.62 at SRS)				86	
2,960.5 (770.5 at ANL-W, 2,190 at SRS)				828		1,178.5 (770.5 at ANL-W, 408 at SRS)				924	
100.7 (10.7 at ANL-W, 90 at SRS)				12.8		23.2 (6.7 at ANL-W, 16.5 at SRS)				14.1	
Person-rem	***LCF***			***Person-rem***	***LCF***	***Person-rem***	***LCF***			***Person-rem***	***LCF***
0.013	6.1×10^{-6}			0.00035	1.7×10^{-7}	0.013	6.1×10^{-6}			0.00035	1.7×10^{-7}
0.002	7.9×10^{-7}			4.7×10^{-5}	1.9×10^{-8}	0.002	7.9×10^{-7}			4.7×10^{-5}	1.9×10^{-8}
3.0×10^{-6}	1.5×10^{-9}			1.1×10^{-12}	1.0×10^{-15}	3.0×10^{-6}	1.5×10^{-9}			1.1×10^{-12}	1.0×10^{-15}

ERPG = Emergency Response Planning Guideline, LCF = latent cancer fatalities, MEI = maximally exposed individual
DBA = design-basis accident, BDBA = beyond-design-basis accident, SNF = spent nuclear fuel

[a] Only occurs for the initial two years during fuel packaging and handling, and Electrometallurgical Treatment Demonstration Project waste stabilization.

[b] Over a period of six months.

[c] Population doses (population and total worker) are in person-rem per year; individual doses are in millirem.

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

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

2.11 REFERENCES

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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 environments of the Idaho National Engineering and Environmental Laboratory and 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 Council on Environmental Quality's Guidance under National Environmental Policy Act (NEPA) regulations (40 CFR 1500-1508) 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 alternatives.

The affected environment for each candidate site presented in this section is based on the *Surplus Plutonium Disposition Draft Environmental Impact Statement* (DOE 1998b), unless otherwise noted. Additional information on the affected environment was determined from other recent environmental impact statements, 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 National Environmental Policy Act (NEPA) documents such as the *Idaho National Engineering and Environmental Laboratory Advanced Mixed Waste Treatment Final Environmental Impact Statement* (DOE 1999a) and the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement* (DOE 1998f).

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 approximately 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 reactor research, other INEEL facilities are operated to support reactor operations. These facilities include high-level radioactive 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 (such as roads). Generalized land uses at INEEL and the vicinity are shown in **Figure 3–1.** Facility operations include industrial and support operations associated with energy research and waste management activities. 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. 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. DOE land-use plans and policies applicable to INEEL are discussed in the *DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (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 (LMITC 1997). Five nuclear test reactors, including the Experimental 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 Hot Fuel Examination facility are also located at the site (DOE 1996a).

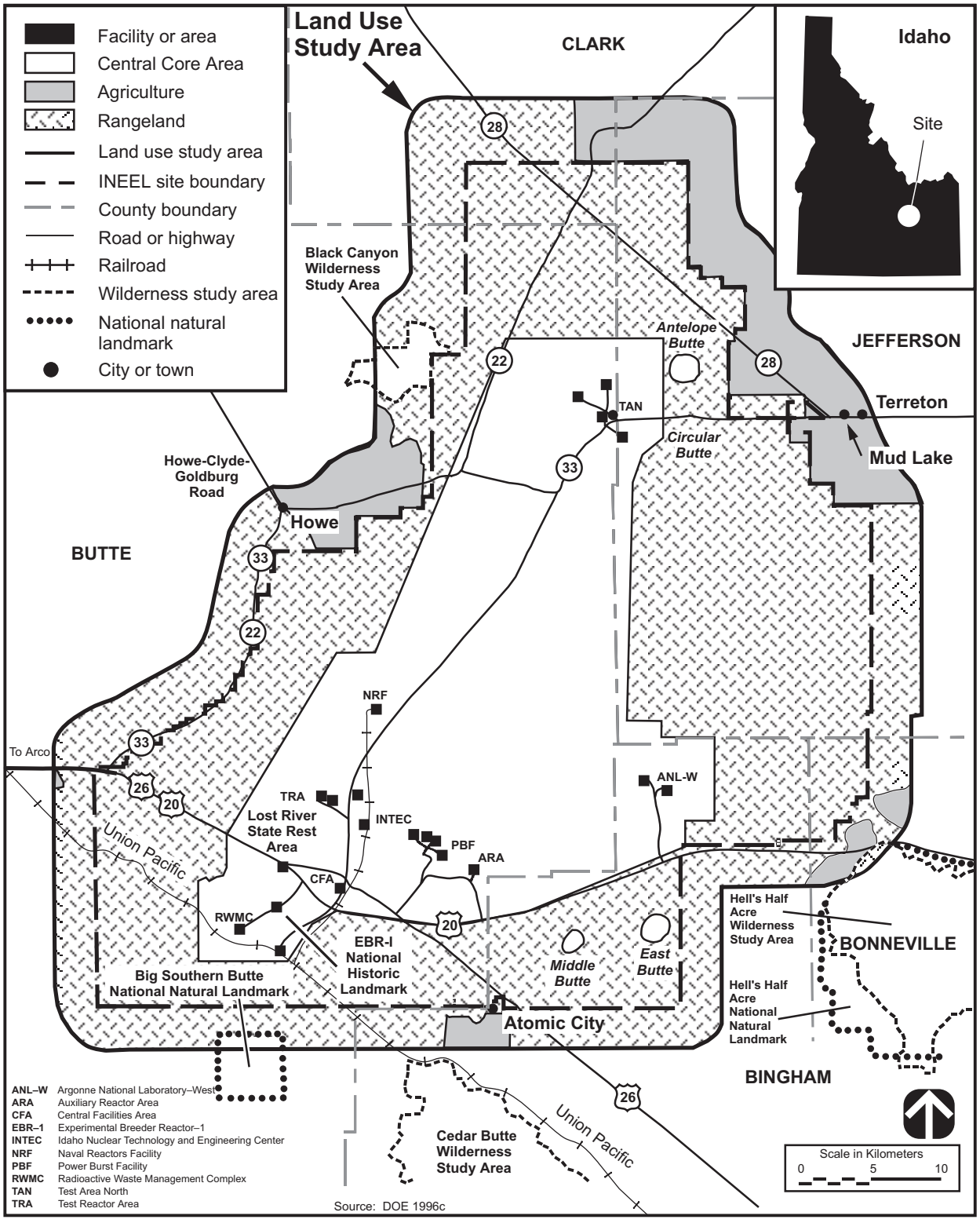

Figure 3–1 Generalized Land Use at INEEL and Vicinity

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 are under Bureau of Land Management jurisdiction and are designated as Visual Resource Management Class II areas. INEEL itself generally consists of open desert land mostly covered by large sagebrush and grasslands. Most 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 master plan, 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 designation 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, Experimental Breeder Reactor-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 Big Lost River at 19 kilometers (11.8 miles), Big Southern Butte National Natural Landmark at 30 kilometers (18.6 miles), East Butte at 9 kilometers (5.6 miles), Middle Butte at 11 kilometers (6.8 miles), Hell's Half Acre Wilderness Study Area, and Hell's Half Acre National Natural Landmark at 15 kilometers (9.3 miles).

3.2.2 Site Infrastructure

Site infrastructure includes those utilities and other resources required to support construction 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**.

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)	6,100,000,000	43,000,000,000

[a] Includes paved and unpaved roads.
[b] FY 1997 data based on INEEL 1998.
[c] As supplies get low, more can be supplied by truck or rail.
Source: DOE 1998a, except as noted in footnote b.

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 by truck and train. About 140 kilometers (87 miles) of paved surface has 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 the 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.

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).

3.2.2.3 Fuel

Fuels consumed at INEEL include several liquid petroleum fuels, coal, and propane gas. All fuels are transported to the site for storage and use. Fuel storage is provided for each facility, and the inventories are restocked as necessary. The current site usage of fuel oil is about 5.7 million liters per year (1.5 million gallons per year). The current site usage of coal is about 11,340 metric tons per year (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 per year (11 billion gallons per year) for the site. The current site usage is about 6.1 billion liters per year (1.6 billion gallons per year).

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).

INEEL is within Eastern Idaho Intrastate Air Quality Control Region # 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 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, coal piles, industrial processes, stationary diesel engines, vehicles, and fugitive dust from burial and construction activities. The existing ambient air concentrations attributable to sources at INEEL are presented in Table 3–2. These concentrations are based on dispersion modeling using maximum emissions for the year 1990 and meteorological data from 1992, and are expected to bound the actual INEEL contribution to ambient levels. Only those toxic and hazardous air pollutants that would be emitted for any of the alternatives evaluated in this EIS are presented. Concentrations attributable to INEEL are in compliance with applicable guidelines and regulations (Table 3–2).

The nearest Prevention of Significant Deterioration Class I area[1] to INEEL is Craters of the Moon Wilderness Are, 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.

[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 3–2 Comparison of Modeled Ambient Air Concentrations From INEEL Sources With Most Stringent Applicable Standards or Guidelines

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*[a]	*INEEL Concentration*[g] *(micrograms per cubic meter)*
Criteria pollutants			
Carbon monoxide	8 hours 1 hour	10,000[b] 40,000[b]	284 614
Nitrogen dioxide	Annual	100[b]	4
Ozone	8 hours	157[c]	(d)
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50[b] 150[b] 150[c]	3 33 (e)
$PM_{2.5}$	3 year annual 24 hours (98^{th} percentile over 3 years)	15[c] 65[c]	(e) (e)
Sulfur dioxide	Annual 24 hours 3 hours	80[b] 365[b] 1,300[b]	6 135 579
Hazardous and other toxic compounds[f]			

[a] The more stringent of the Federal and state standards are presented if both exist for the averaging period.
[b] Federal and state standard.
[c] Federal standard.
[d] Not directly emitted or monitored by the site.
[e] No data are available with which to assess particulate matter concentrations.
[f] Any hazardous and toxic compounds would be well below regulatory levels (ANL 1999b).
[g] Concentrations based on 1990 emissions and 1992 meteorological data.
Source: 40 CFR 50, ID DHW 1998, Moor and Peterson 1999, 62 FR 38855, 62 FR 38652.

Prevention of Significant Deterioration permits have been obtained for the coal-fired steam-generating facility (located next to the Idaho Nuclear Technology and Engineering Center) and the Fuel Processing Facility. The Fuel Processing Facility is not expected to be operated (DOE 1996c). In addition to this facility, 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.

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 60 kilometers [19.3 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).

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

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

Pollutant	*Averaging Time*	*Allowable Prevention of Significant Deterioration increment[a] (micrograms per cubic meter)*	*Amount of Prevention of Significant Deterioration increment consumed (micrograms per cubic meter)*	*Percent of Prevention of Significant Deterioration Increment Consumed*
Nitrogen dioxide [b]	Annual	2.5	0.004	1.8
Respirable particulates [c]	Annual 24hours	4 8	0.008 0.6	0.2 7.5
Sulfur dioxide	Annual 24 hours 3 hours	2 5 25	0.09 1.8 5.9	4.5 36 24

[a] All increments specified are State of Idaho standards (ID DHW 1998).
[b] Assumes that the New Waste Calcining Facility (the largest source of nitrogen dioxide emissions at INEEL) operates for the entire year.
[c] Data on particulate size are not available for most sources. For purposes of comparison to the respirable particulate increments, it is conservatively assumed that all particulates emitted are of respirable size (i.e., 10 microns or less in diameter).
Source: DOE 1999a.

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

Pollutant	*Averaging Time*	*Allowable Prevention of Significant Deterioration increment [a] (micrograms per cubic meter)*	*Amount of Prevention of Significant Deterioration increment consumed (micrograms per cubic meter)*	*Percent of Prevention of Significant Deterioration Increment Consumed*
Nitrogen dioxide [b]	Annual	25	1.4	5.7
Respirable particulates [c]	Annual 24 hours	17 30	0.92 15	5.4 51
Sulfur dioxide	Annual 24 hours 3 hours	20 91 512	2.4 29 132	12 32 26

[a] All increments specified are State of Idaho standards (ID DHW 1998).
[b] Assumes that the New Waste Calcining Facility operates for the entire year.
[c] Data on particulate size are not available for most sources. For purposes of comparison to the respirable particulate increments, it is conservatively assumed that all particulates emitted are of respirable size (i.e., 10 microns or less in diameter).
Source: DOE 1999a.

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 point 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).

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 due to 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 1998b). The noise generated at INEEL is not propogated at detectable levels offsite, 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. 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 study of the 100-year peak flow for the Big Lost River has been completed by the U.S. Geological Survey (USGS 1998). The 100-year and 500-year flood plains are being studied by the Bureau of Reclamation. No flood maps of the Big Lost River are available from the Federal Emergency Management Agency or other agencies (Abbott, Crockett, and Moor 1997). Flood diversion facilities constructed in 1958 and enlarged in 1984 secured INEEL from the 300-year flood (DOE 1996c).

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 failure of Mackay Dam on the Big Lost River indicate that flood waters would reach ANL-W (**Figure 3–3**).

ANL-W discharges 11,900,000 liters per year (3,140,000 gallons per year) of nonhazardous liquid waste to the sewage pond and 68,000,000 liters per year (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).

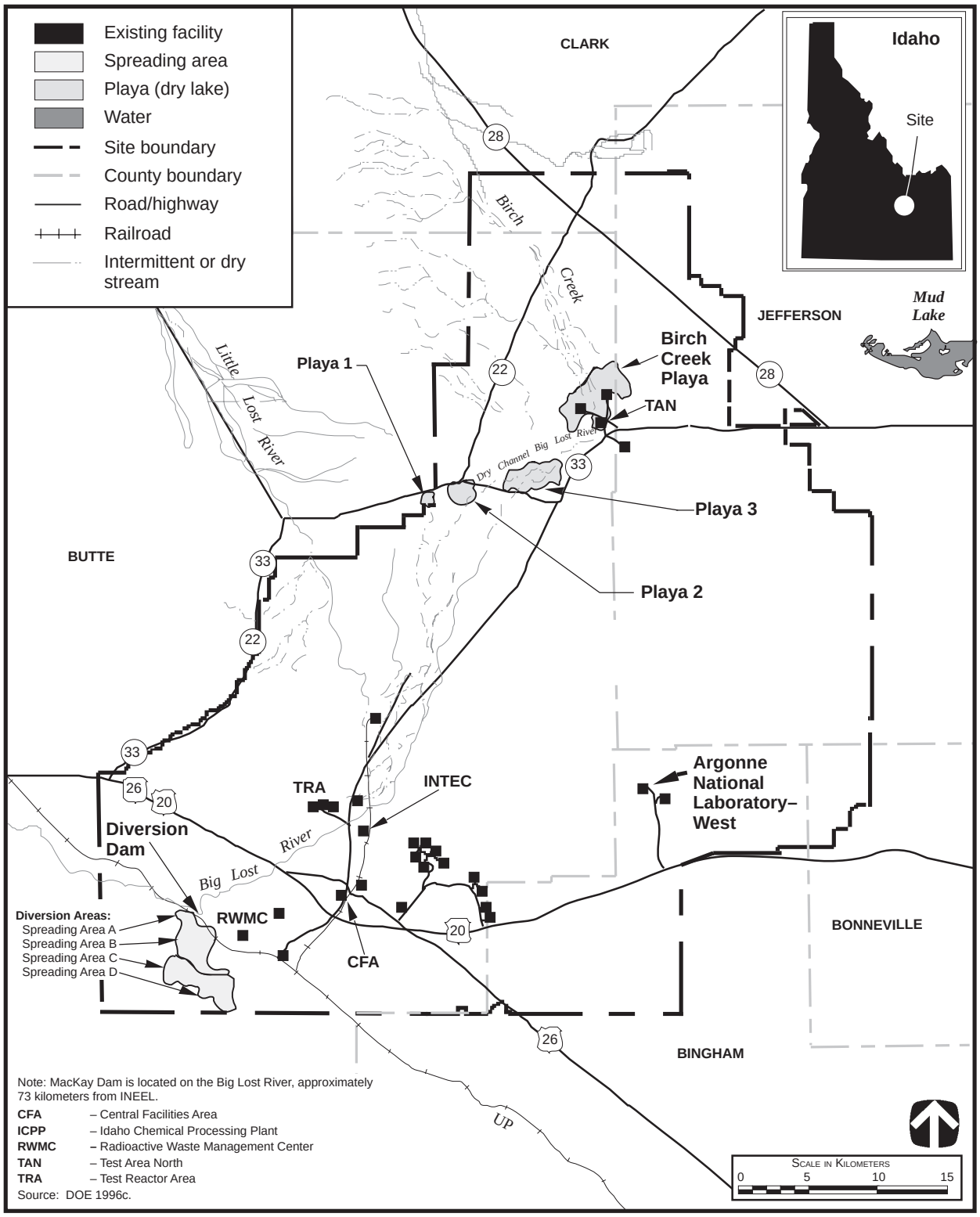

Figure 3–2 Surface Water Features at Idaho National Engineering and Environmental Laboratory

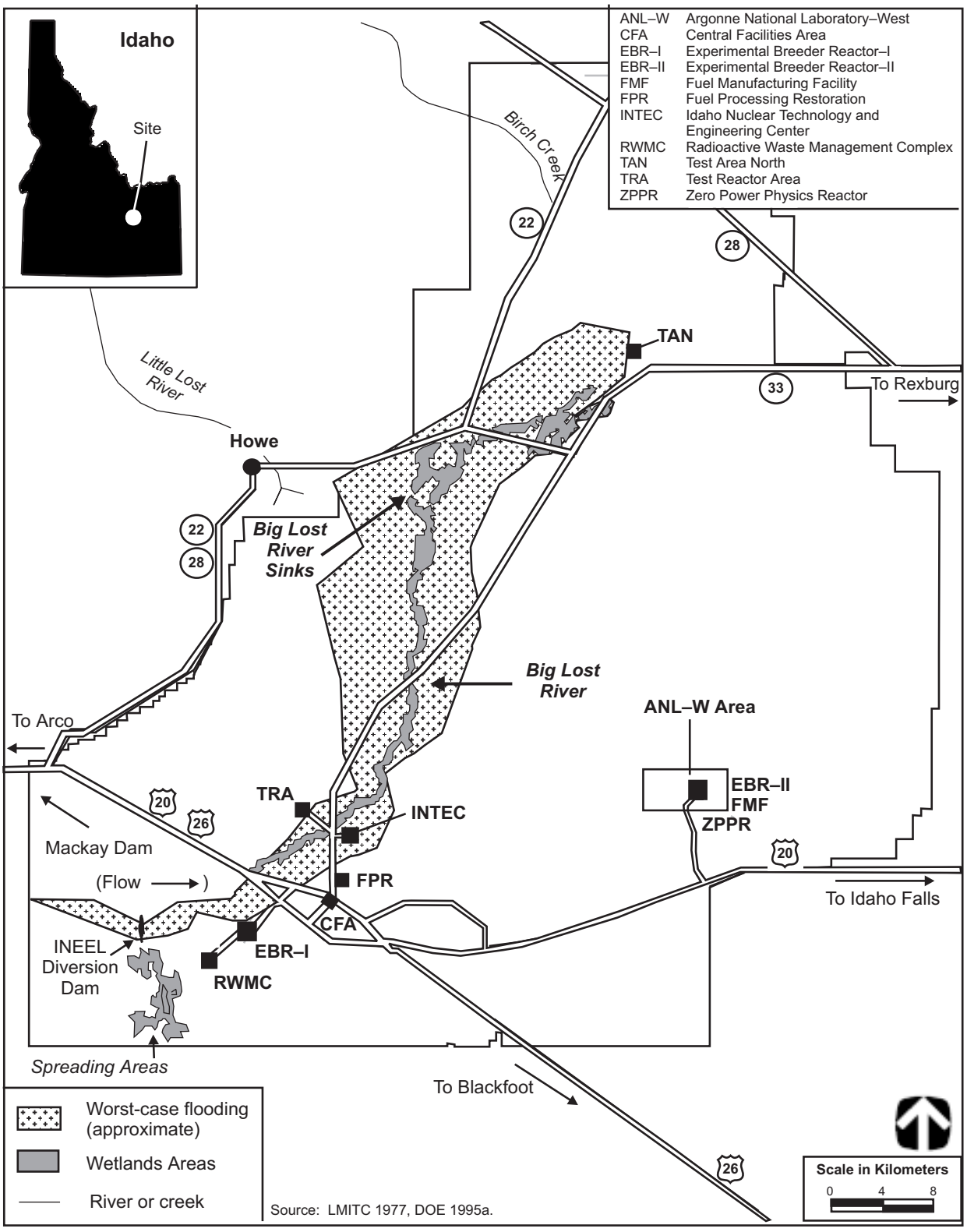

Figure 3–3 Flood Area for the Probable Maximum Flood-Induced Overtopping Failure of the Mackay Dam

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 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 (DOE 1996c).

Groundwater generally flows laterally at a rate of 1.5 to 6.1 meters per day (5 to 20 feet per day). It emerges in springs along the Snake River from Milner to Bliss, Idaho. Depth to the groundwater table ranges from about 61 meters (200 feet) below ground in the northeast corner of the site to about 305 meters (1,000 feet) in the southeast corner (DOE 1995a, DOE 1996c). Perched water tables (i.e., bodies of groundwater lying above a more extensive aquifer) occur below the site. These perched water tables tend to slow the migration of pollutants that might otherwise reach the Snake River Plain aquifer (DOE 1996c).

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 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 due to 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. Components of nonradioactive waste 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 (DOE 1996c).

INEEL uses about 7.2 billion liters per year (1.9 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 per second (80 cubic feet per second) with a maximum water consumption of 42 billion liters per year (11 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 to 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 per year (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

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. There is no potential for sinkholes at INEEL. Lava tubes, which could have adverse effects similar to those of sinkholes, do occur in the INEEL area.

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 the Idaho Nuclear Technology and Engineering Center (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.

According to the Uniform Building Code, INEEL, located on the Eastern Snake River Plain, is in Seismic Zone 2B, meaning that moderate damage could occur as a result of an earthquake. No earthquakes have been recorded within 48 kilometers (30 miles) of the site (DOE 1998b). 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.

Volcanic hazards at INEEL can come from sources inside or outside the Snake River Plain. Most of the basaltic volcanic activity occurred at the Craters of the Moon National Monument 20 kilometers (12 miles) southwest of INEEL between 4 million and 2,100 years ago. The probability of volcanic activity affecting facilities at INEEL is very low. A detailed discussion relating to the probability of volcanism affecting INEEL is presented in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c).

Four basic soilscapes exist on the 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 (DOE 1997b). 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 1998b).

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 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 important habitat for species native to the region. 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 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.

The 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 (LMITC 1997). Numerous raptors, such as the golden eagle and prairie falcon, and carnivores, such as the coyote and mountain lion, are also 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 Flack 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).

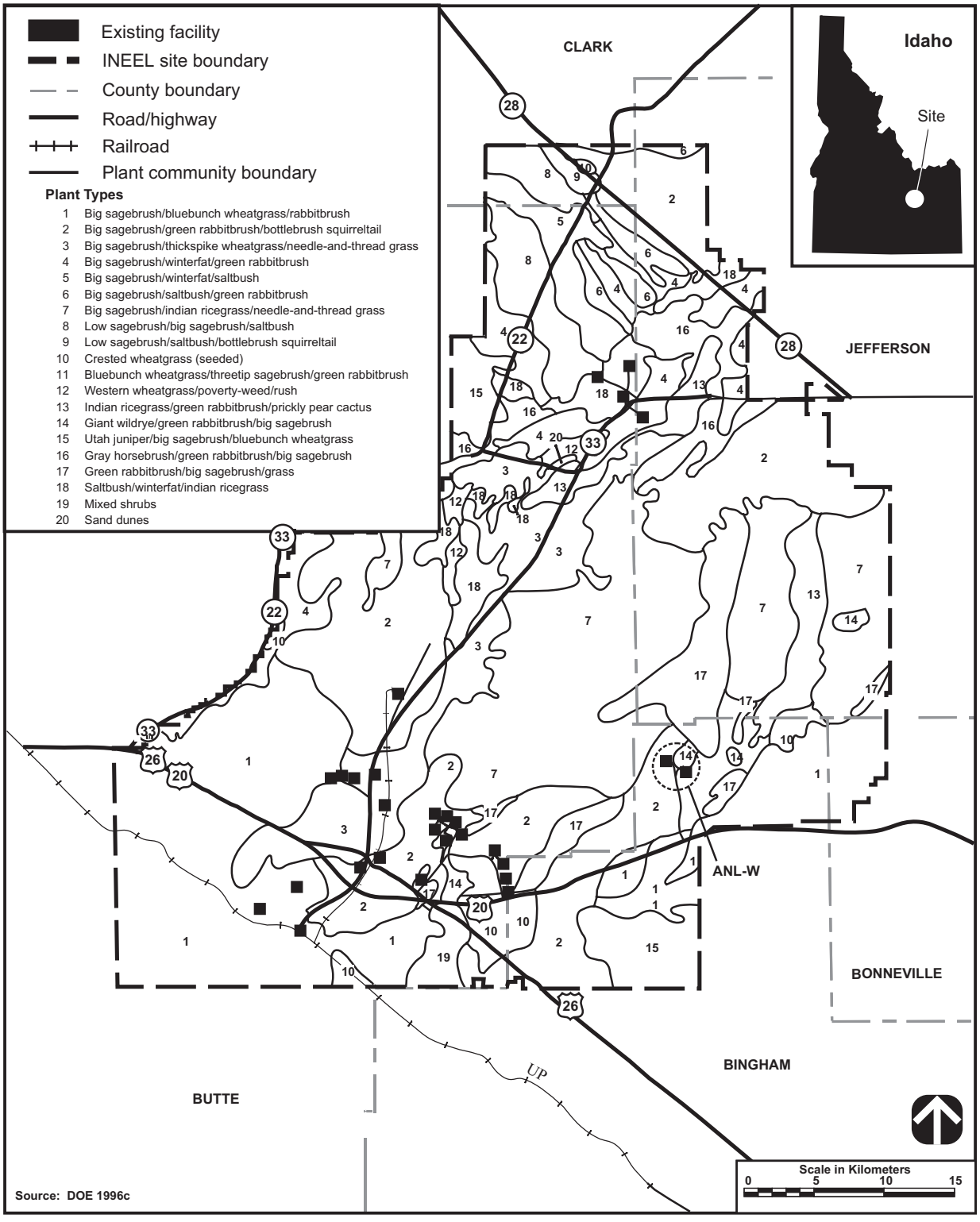

Figure 3–4 Distribution of Plant Communities at Idaho National Engineering and Environmental Laboratory

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 due to recent years of only intermittent flows. The Big Lost River spreading areas and Big Lost River sinks are seasonal wetlands and are located 33 kilometers (21 miles) west southwest and 23 kilometers (14 miles) northeast 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 Flack 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, the bald eagle and peregrine falcon, are Federally listed as threatened and endangered, respectively. Each is listed as endangered by Idaho. The bald eagle rarely has been seen in the western and northern portion 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 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 initiated 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 formally evaluated 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 included items such as pieces of Shoshone brownware pottery, projectile points, farm implements, and broken glass. The archaeological sites included projectile points, scrappers, and volcanic glass flakes. A number of recent items such as a belt buckle and a large scatter of cans also were found. 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 (DOE 1996a). 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. Experimental Breeder Reactor-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.

As noted under Prehistoric Resources above, a limited number of recent artifacts have been located in the vicinity of ANL-W. The Experimental Breeder Reactor-II has been designated as an American Nuclear Society Historical Landmark (DOE 1997c). Consultation has been initiated 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 initiated 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 geological 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 floodplain 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 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 (DOE 1998b). 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 four-county area in Idaho in which 94.4 percent of all INEEL employees reside (**Table 3–5**). In 1997, total INEEL employment was 8,291 persons (5.5 percent of the regional economic area civilian labor force).

Table 3–5 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 1998b.

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 150,835. 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 1998b). 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 2.0 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 thresholds. 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 the 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 (15.0 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 3–6**. 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 3–6 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]
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 3–7.** 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.

Table 3–7 Radiation Doses to the Public From Normal INEEL Operations in 1997 (Total Effective Dose Equivalent)

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

Using a risk estimator of 500 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.)

According to the same risk estimator, 1.2×10^{-4} excess 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 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 1.2×10^{-4} fatal cancers estimated from INEEL operations in 1997.

INEEL workers receive the same doses 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 3–8.** These doses fall within the radiological regulatory limits of 10 CFR 835 (DOE 1995a). According to a risk estimator of 400 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 workforce of the more radiosensitive infant and child age groups.

Table 3–8 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. It has therefore established an administrative control level of 2,000 millirem per year; 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.

Source: DOE 1995a, DOE 1998g.

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) are also 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 difference 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 at ANL-W are 6.0×10^{-16} microcuries per milliliter and 2.0×10^{-14} microcuries per milliliter, respectively. These alpha and beta concentrations are essentially the same as those measured at offsite control locations (Evans et al. 1998).

3.2.10.2 Chemical Environment

The background chemical environment important to human health consists of the atmosphere, which may contain hazardous chemicals that can be inhaled; drinking water, which may contain hazardous chemicals that can be ingested; and other environmental media through which people may come in contact with hazardous chemicals (e.g., surface water during swimming, soil through direct contact, or food). Hazardous chemicals can cause cancerous and noncancerous health effects.

Effective administrative and design controls that decrease hazardous chemical releases to the environment and help achieve compliance with permit requirements (e.g., air emissions and National Pollutant Discharge Elimination System [NPDES] permit requirements) contribute to minimizing health impacts on the public. The effectiveness of these controls is verified through the use of monitoring information and inspection of mitigation measures. Health impacts on the public may occur by inhaling air containing hazardous chemicals released to the atmosphere during normal INEEL operations. Risks to public health from other possible pathways, such as ingesting contaminated drinking water or direct exposure, are lower than those via the inhalation pathway. At INEEL, the risk to public health from water ingestion and direct exposure pathways is low because surface water is not used for drinking or as a receptor for wastewater discharges.

The baseline concentrations are estimates of the highest existing offsite concentrations and represent the highest concentrations to which members of the public could be exposed. These concentrations are in compliance with applicable guidelines and regulations. Information on estimating the health impacts of hazardous chemicals is presented in Appendix E.

Exposure pathways to INEEL workers during normal operation may include inhaling contaminants in the workplace and direct contact with hazardous materials. The potential for health impacts varies among facilities and workers, and available information is insufficient for a meaningful estimate of impacts. However, workers are protected from workplace hazards through appropriate training, protective equipment, monitoring, substitution, and engineering and management controls. INEEL workers also are protected by adherence to Occupational Health and Safety Administration (OSHA) and EPA standards that limit workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. Appropriate monitoring that reflects the frequency and amounts of chemicals used in operational processes ensures that these standards are not exceeded. Additionally, DOE requires that conditions 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.

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 EIS* (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 workforce 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 workforce 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 EIS* (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 waste, transuranic waste, mixed transuranic, low-level radioactive waste, mixed waste, hazardous, and nonhazardous. Waste generation rates and the inventory of stored waste from activities at INEEL are provided in **Table 3–9.** The INEEL waste management capabilities are summarized in **Table 3–10.** 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 *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995a).

EPA placed INEEL on the National Priorities List[1] on December 21, 1989. In accordance with CERCLA, DOE entered into a consent order with 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.

[1]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 3–9 Waste Generation Rates and Inventories at INEEL

Waste Type	*Generation Rate (cubic meters)*	*Inventory (cubic meters)*
High-Level Radioactive	0[a]	4,000[b]
Transuranic	0[c]	65,000[d]
Low-Level Radioactive	6,400[e]	6,000[f]
Mixed	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.
[c] DOE 1995a.
[e] LMITC 1998.
[f] Bright 1999.
[g] DOE 1998e.
[h] Includes 760 cubic meters that are recyclable.
[I] Generally, hazardous and nonhazardous wastes are not held in long-term storage.
[j] Projected annual average generation amounts for 1997–2006.
Note: To convert from cubic meters to cubic yards, multiply by 1.31.
Sources: Given in footnotes b through g, above.

Table 3–10 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	Online			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	Online	X						
ANL–W Remote Treatment Facility	42	Planned for 2000		X	X	X	X		
ANL–W Hot Fuel Examination Facility Waste Characterization Area	37	Online		X	X				
INTEC Waste Immobilization Facility	48	Planned for 2020			X	X	X		
INTEC Liquid Effluent Treatment and Disposal Facility	11,365	Online					X		
INTEC High-Level Radioactive Waste Evaporator	6,138	Online			X	X	X		
INTEC Process Equipment Waste Evaporator	13,000	Online			X	X	X		
ANL–W Sodium Processing Facility	698	Online					X		

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed	Haz	Non-Haz
Test Area North Cask Dismantlement	11	Online					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	Online					X		
Waste Experimental Reduction Facility	49,610	Online				X	X	X	
INTEC Cold Waste Handling Facility	3,700	Online							X
INTEC Sewage Treatment Plant	3,200,000	Online							X
Storage Facility (cubic meters)									
INTEC Tank Farm	12,533	Online			X		X		
INTEC Calcine Bin Sets	6,950	Online	X						
ANL–W Radioactive Sodium Storage	75	Online			X		X		
ANL–W Sodium Components Maintenance Shop	200	Online					X		
ANL–W Radioactive Scrap and Waste Storage	193	Online		X	X	X	X		
ANL–W EBR-II Sodium Boiler Drain Tank	64	Online					X		
ANL–W Hot Fuel Examination Facility Waste Characterization Area	37	Online		X	X				
INTEC Fluorinel Dissolution Process High-Efficiency Particulate Air Storage	25	Online			X		X		
INTEC New Waste Calcining Facility High-Efficiency Particulate Air Storage	56	Online			X		X		
INTEC Chemical Processing Plant-1619 Storage	45	Online					X	X	
INTEC Chemical Processing Plant-1617 Staging	8,523	Online					X	X	
Radioactive Waste Management Complex Transuranic Storage Area-RE[a]	64,900	Online		X	X	X	X		
Radioactive Waste Management Complex Waste Storage[a]	112,400	Online		X	X	X	X		
Radioactive Waste Management Complex Intermediate-Level Storage	100	Online		X					
Waste Reduction Operations Complex Power Burst Facility Mixed Waste Storage	129	Online					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	Online					X	X	
Power Burst Facility Waste Experimental Reduction Facility Waste Storage Building	685	Online					X	X	
Test Area North 647 Waste Storage	104	Online					X	X	
Test Area North 628 Specific Manufacturing Complex Container Storage	125	Online					X	X	
Disposal Facility (cubic meters per year)									
Radioactive Waste Management Complex Disposal Facility	37,700	Online				X			
Central Facilities Area Landfill Complex	48,000	Online							X
Percolation Ponds	2,000,000	Online							X

EBR = Experimental Breeder Reactor, HAZ = hazardous waste, HLW = high-level radioactive waste, INTEC = Idaho Nuclear Technology and Engineering Center, LLW = low-level radioactive waste, TRU = transuranic waste

[a] 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 per gram).

Source: DOE 1998b, DOE 1999d.

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-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 metallic 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 met 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 Facility Compliance Act.

3.2.11.3 Transuranic Waste

Transuranic waste generated since 1972 is segregated into contact-handled and remotely handled categories and stored at the Radioactive Waste Management Complex in a form designed for eventual retrieval (DOE 1996c). Some transuranic waste is also 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 is currently 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 1999c). 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 (INEEL 1999a). 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. Construction of a mixed waste Disposal Facility and Plasma Hearth Treatment Facility is being considered to support commercial treatment of mixed transuranic waste and alpha-contaminated mixed waste subject to funding restraints and additional NEPA review (DOE 1998b).

Waste containing between 10 and 100 nanocuries per gram of transuranic radionuclides is called alpha low-level radioactive waste. Although this waste is technically 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 these wastes will be treated by the Advanced Mixed Waste Treatment Project and then disposed of at the Waste Isolation Pilot Plant (DOE 1998b).

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 1998b). 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 offsite 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 1998b).

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 wastes are 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 1998b).

As part of the site treatment plans required by the Federal Facility 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 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 Reductions Operations Complex Macroencapsulation (March 1999); (7) Waste Reduction Operations Complex Mercury Retort (March 2000); (8) Debris Treatment (September 2000); and (9) Advanced Mixed Waste Treatment Project (March 2003). Commercial treatment facilities are also being considered, as appropriate. Currently, limited amounts of mixed waste are disposed of at Envirocare of Utah (DOE 1998b).

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 offsite for treatment and disposal (DOE 1995a). Offsite shipments are surveyed to determine that the wastes have 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 1998b). 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).

The Cold Waste Handling Facility was recently put into operation at INTEC. This system allows increased volumes of nonhazardous waste to be inspected, recycled, shredded, compacted, and segregated, thereby reducing the amount of material sent to disposal. Combustible waste is taken to the solid waste handling facility for sorting and cubing. The cubed material is taken to a steam-generating facility and converted from waste to energy (DOE 1998b).

Sewage is disposed of in surface impoundments in accordance with terms of the October 7, 1992, consent order. Waste 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 per day (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 per year (4,190,000 cubic yards per year) (DOE 1998b).

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 1998d).

The INEEL waste minimization program has significantly reduced 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, 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 1998d). During 1993–1995, INEEL recycled more than 680,400 kilograms (1.5 million pounds) of paper and cardboard (DOE 1998b). 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 Environmental Impact Statement 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 Hazardous Waste (Waste Management Programmatic EIS) (DOE 1997a) are shown in **Table 3–11** for the waste types analyzed in this EIS. A decision on the future management of these wastes could result in the construction of new waste management facilities at INEEL and the closure of other facilities. Decisions on the various waste types are expected to be announced in a series of Record of Decisions to be issued on the Waste Management Programmatic EIS. The transuranic waste Record of Decision was issued on January 20, 1998 (63 FR 3629), and the hazardous waste Record of Decision on August 5, 1998 (63 FR 41810). The transuranic waste Record of Decision states, "each of the Department's sites that currently has or will generate 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. More detailed information and DOE's alternatives for the future configuration of waste management facilities at INEEL are presented in the Waste Management Programmatic EIS and the hazardous waste and transuranic waste Record of Decisions.

Table 3–11 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 prefers to treat INEEL's low-level radioactive waste on site. INEEL could be selected as one of the regional disposal sites for low-level radioactive waste.[a]
Mixed	DOE prefers regionalized treatment of mixed waste at INEEL. This includes the onsite treatment of INEEL's wastes and could include treatment of some mixed waste generated at other sites. INEEL could be selected as one of the regional disposal sites for mixed waste.[a]
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.[c]

[a] From the Waste Management Programmatic EIS (DOE 1997a).
[b] From the ROD for transuranic waste (63 FR 3629).
[c] From the ROD for hazardous waste (63 FR 41810).
Source: 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 operation 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 separations 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 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.

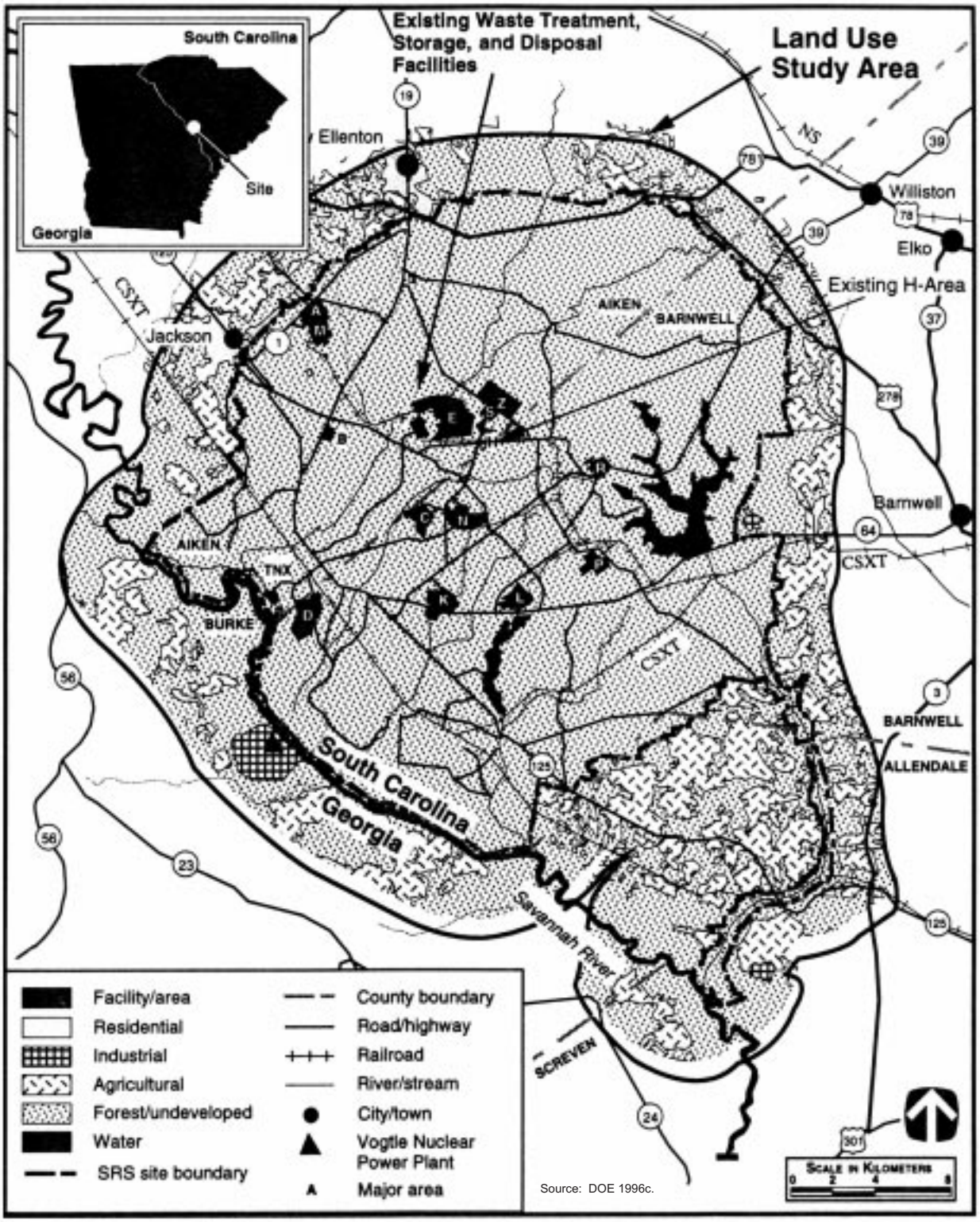

Figure 3–5 Generalized Land Use at Savannah River Site and Vicinity

Outdoor public recreation facilities are plentiful and varied in the SRS region. Included are the Sumter National Forest, 75 kilometers (47 miles) to the northwest; Santee National Wildlife Refuge, 80 kilometers (50 miles) to the east; and Clarks Hill/Strom Thurmond Reservoir, 70 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 Wildlife Management Area, which occupies over 1,930 hectares (4,770 acres) of SRS adjacent to the Savannah River, is open to the public for hunting and fishing.

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) (i.e., 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 primarily managed 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 have historically 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 1998f). This facility would be used for processing sodium-bonded spent nuclear fuel if the melt and dilute alternative were selected.

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 designation 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 designation from Class II to Class III. Management activities within these classes may be seen, but should not dominate the view.

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 designation.

3.3.2 Site Infrastructure

Site infrastructure includes those utilities and other resources required to support construction 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 3–12.**

Table 3–12 Savannah River Site-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] As supplies get low, more can be supplied by truck or rail.
Source: DOE 1998b.

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 Sub-Region, 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, 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 number 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-liters per day (4.5-million-gallons per day) water treatment plant in A-Area; two wells and an 8.3-million-liters per day (2.2-million-gallons 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 liters per minute (444 gallons 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 is frequently 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 1994b).

SRS is near the center of the Augusta-Aiken Interstate Air Quality Control Region #53. None of the areas within SRS and its surrounding counties are designated as nonattainment areas with respect to the National Ambient Air Quality Standards (NAAQS) for criteria air pollutants (40 CFR 50). Applicable NAAQS and state ambient air quality standards are presented in **Table 3–13.**

Table 3–13 Comparison of Modeled Ambient Air Concentrations From Savannah River Site Sources With Most Stringent Applicable Standards or Guidelines

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*[a]	*Savannah River Site Concentration (micrograms per cubic meter)*
Criteria pollutants			
Carbon monoxide	8 hours 1 hour	10,000[b] 40,000[b]	632 5010
Nitrogen dioxide	Annual	100[b]	8.8
Ozone	8 hours	157[c]	(d)
PM_{10}	Annual 24 hours (interim) 24 hours (99th percentile over 3 years)	50[b] 150[b] 150[c]	4.8 80.6 (e)
$PM_{2.5}$	3 year annual 24 hours (98th percentile over 3 years)	15[c] 65[c]	(e) (e)
Sulfur dioxide	Annual 24 hours 3 hours	80[b] 365[b] 1,300[b]	16.3 215 690
State regulated pollutants			
Gaseous fluoride	30 days 7 days 24 hours 12 hours	0.8[f] 1.6[f] 2.9[f] 3.7[f]	0.11 0.06 1.2 2.4
Total suspended particulates	Annual	75[f]	43.3
Hazardous and other toxic compounds			
1,1,1-Trichloroethane	24 hours	9,550	22
Benzene	24 hours	150	31
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	less than 0.01
Methyl alcohol	24 hours	1,310	0.51
Methyl ethyl ketone	24 hours	14,750	0.99
Methyl isobutyl ketone	24 hours	2,050	0.51
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 0.001

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*[a]	*Savannah River Site Concentration (micrograms per cubic meter)*
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

[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] Federal standard.
[d] Not directly emitted or monitored by the site.
[e] No data is available with which to assess particulate matter concentrations.
[f] South Carolina state standard.
Source: DOE 1998f, Bickford et al. 1997, SCDHEC 1998, 40 CFR 50, 62 FR 38855, 62 FR 38652.

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 3–13 presents the ambient air concentrations attributable to sources at SRS. These concentrations are based on dispersion modeling using emissions for the year 1994 (DOE 1998f). 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 3–13 that are attributable to SRS are in compliance with applicable guidelines and regulations.

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, lead, ozone, 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 the F-Area and L-Area are diesel generators.

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 operations 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 1-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 1-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 Creek, Beaver Dam Creek, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs Creek. Upper Three Runs Creek has 2 tributaries, Tims Branch and Tinker Creek; Pen Branch has 1 tributary, called Indian Grave Branch; and Steel Creek has 1 tributary, called Meyers Branch **(Figure 3–6)** (DOE 1996c).

There are two manmade lakes at SRS: L-Lake, which discharges to Steel Creek, and Par Pond, which discharges to Lower Three Runs Creek. Also, up to 350 to 400 Carolina bays—i.e., closed depressions capable of holding water—occur throughout the site. While these bays receive no direct effluent discharges, they do receive stormwater runoff (DOE 1996c, DOE 1998f, 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 per year (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 per second (10,000 cubic feet per second). Three large upstream reservoirs, Hartwell, Richard B. Russell, and Strom Thurmond/Clarks Hill, regulate the flow in the Savannah River, thereby lessening the impacts of drought and flooding on users downstream (DOE 1995b).

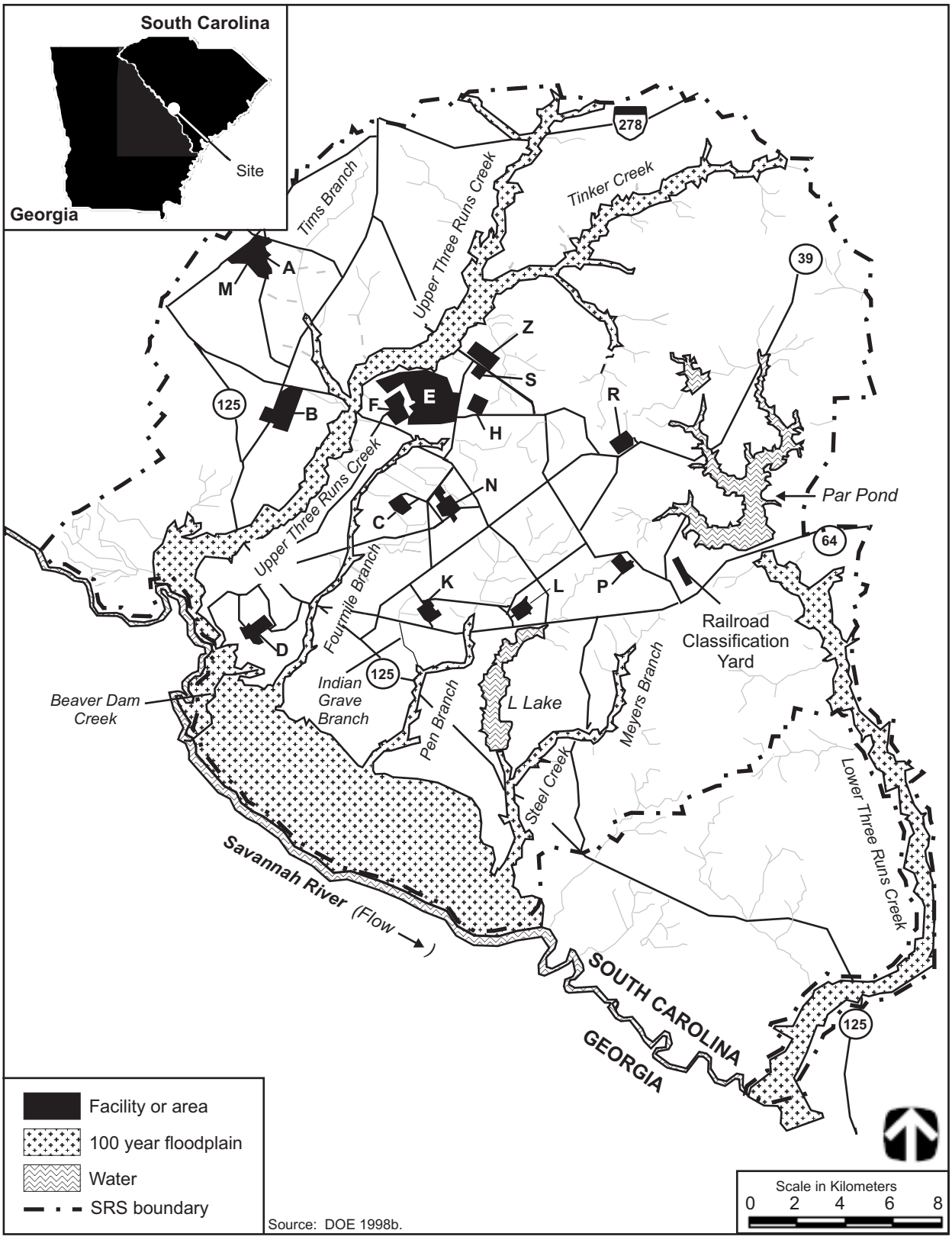

Figure 3–6 Locations of Water Bodies and Floodplains at the Savannah River Site

Several communities in the area use the Savannah River as a source of domestic water. The nearest downstream water intake is the Beaufort-Jasper Water Authority in South Carolina, which withdraws 0.23 cubic meters per second (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 floodplain; there is no information available concerning 500-year floodplains (WSRC 1995). A map showing the 100-year floodplain is presented as Figure 3–6. No Federally designated Wild and Scenic Rivers occur within the site (Barghusen and Feit 1995).

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 (Arnett and Mamatey 1996). A comparison of current and historical data shows that the coliform data are within normal fluctuations for river water in this area. For the different river locations, however, there has been an increase in the number of analyses in which standards were not met. The data for the river's monitoring locations generally met 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. As for radiological constituents, tritium is the predominant radionuclide detected above background levels in the Savannah River (Arnett and Mamatey 1996; DOE 1996c).

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 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 Creek and Fourmile Branch (DOE 1995b). Upper Three Runs Creek is a large, cool blackwater stream that flows into the Savannah River. It drains about 54,390 hectares (134,400 acres), and during water year 1991, had a mean discharge of 6.8 cubic meters per second (240 cubic feet per second) near its mouth. The 7-day, 10-year low flow, which is the lowest flow over any 7 days within any 10-year period, is 2.8 cubic meters per second (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 per second (0.035 cubic feet per second). A comparison with the 7-day, 10-year low flow of 2.8 cubic meters per second (100 cubic feet per second) in Upper Three Runs Creek 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 C-Reactor while it was operating. Since those discharges ceased in 1985, the maximum recorded temperature in the stream has been 31°C (89°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 C-Reactor operation was 11.3 cubic meters per second (400 cubic feet per second); since then flows have averaged 1.8 cubic meters per second (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 152.1. When the Savannah River floods, water from Fourmile Branch flows along the northern boundary of the floodplain 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 effluents 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 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 per second (19.3 cubic feet per second) (DOE 1995b; DOE 1997b).

The headwaters of Steel Creek originate near 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 L-Reactor was operating, Steel Creek received cooling water from L-Reactor, ash basin runoff, nonprocess cooling water, powerhouse wastewater, reactor process effluents, sanitary treatment plant effluents, 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 1998f).

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 (DOE 1996c).

Groundwater in the area is used extensively for domestic and industrial purposes. Most municipal and industrial water supplies in Aiken County are withdrawn from the Cretaceous or water table aquifer, while

small domestic supplies are withdrawn from the Congaree or water table aquifer. In Barnwell and Allendale counties, the Congaree aquifer supplies some municipal users. It is estimated that about 13 billion liters per year (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 the SRS (DOE 1995b). All groundwater at the 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 withdraws more than 5 billion liters per year (1.3 billion gallons per year) of groundwater to support site operations (DOE 1998f). 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 the Congaree aquifer 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 per day (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 Creek and Fourmile Branch. Shallow groundwater in the vicinity of F-Area flows toward Upper Three Runs Creek, 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 21 meters (4 to 68 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 were also 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 1998f).

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 is potentially 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 Draft Environmental Impact Statement* (DOE 1998f). The largest fault is the Pen Branch fault. However, there is no evidence of movement within the last 38 million years along this fault.

SRS is located within Seismic Zone 2, indicating moderate damage could occur as a result of an earthquake. 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 0.0002g). Existing information does not conclusively correlate these earthquakes with any of the known faults on the site (1998a). Historically, two large earthquakes have occurred within 160 kilometers (100 miles) of SRS. The Charleston earthquake of 1886 had an estimated Richter magnitude ranging 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-based earthquake (DOE 1995e).

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 43 amphibian, 58 reptile, 213 bird, and 54 mammal species. 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, but 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.

F-Area is an industrial area situated on an upland plateau between the drainage areas of Upper Three Runs Creek 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 is located along Upper Three Runs Creek 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 1998b). 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 the 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).

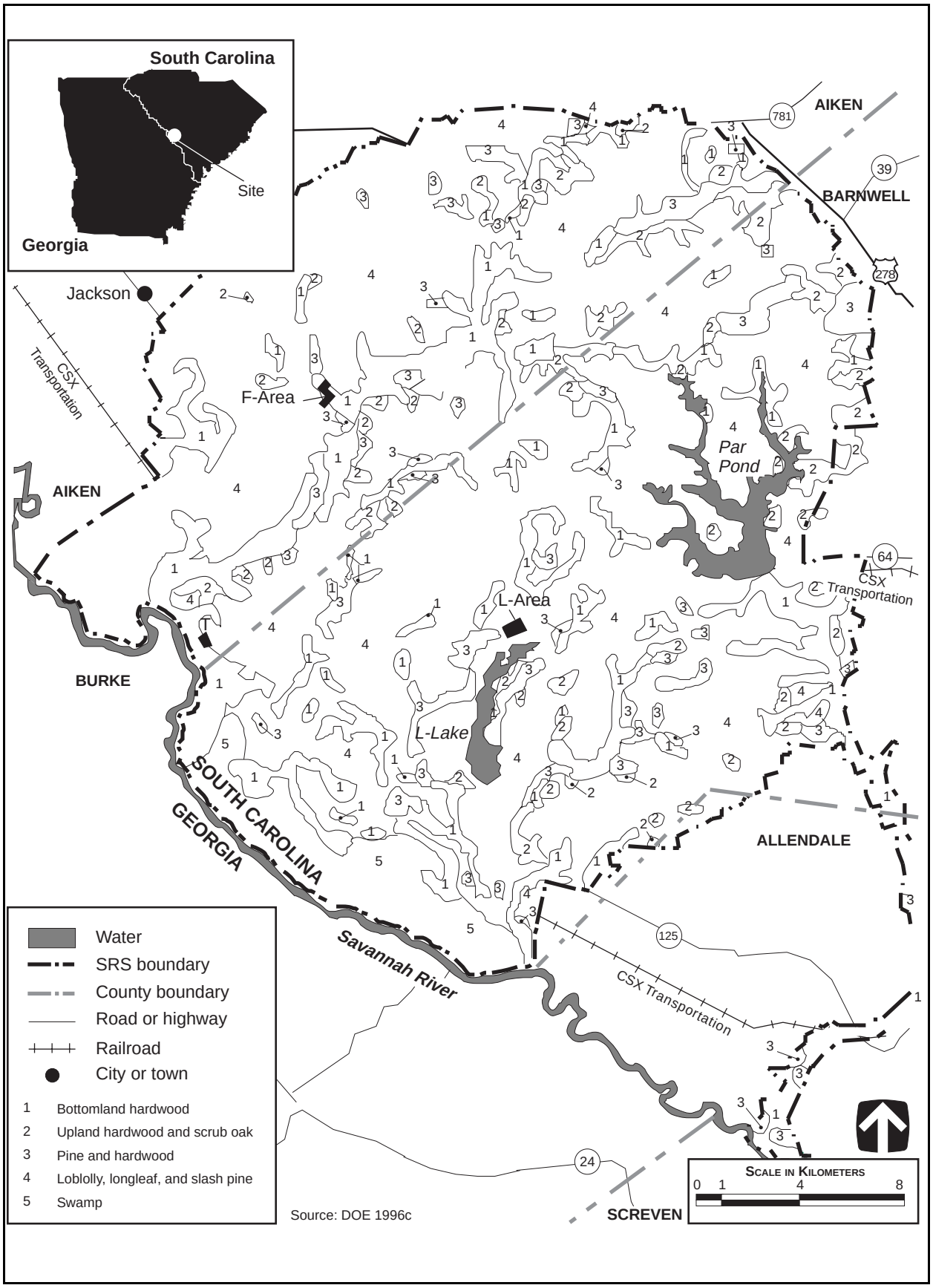

Figure 3–7 Distribution of Plant Communities at Savannah River Site

3.3.6.2 Wetlands

SRS contains approximately 19,800 hectares (49,000 acres) of wetlands, most of which are associated with floodplains, 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 floodplain, 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 1998c). 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 the F-Area are primarily associated with Upper Three Runs Creek 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 primarily classified 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-leaved 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. Although sport and commercial fishing is not allowed on SRS, they do take place on the Savannah River. In the past, water intake structures for 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 the F-Area include Upper Three Runs Creek and Fourmile Branch and their tributaries. Fish species present in Upper Three Runs Creek in the vicinity of the 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 the F-Area include the dusky shiner, creek chubsucker, yellow bullhead, and spotted sunfish. Studies of fish communities in Upper Three Runs Creek and Fourmile Branch indicated that no measurable community level impacts were associated with contaminants from the F-Area seepage basins (DOE 1996b; DOE 1998b).

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 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 L-Reactor, operation of 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 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 the 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 the 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 endangered by South Carolina. Wood storks have been observed 14.5 kilometers (9 miles) from the 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 the 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 the F-Area. The state-listed rare Oconee azalea has been found on steep slopes adjacent to the Upper Three Runs Creek floodplain just northwest of F-Area (DOE 1995b). Consultation has been initiated with both the U.S. Fish and Wildlife Service and the state.

No Federally-listed threatened or endangered species are known to occur in the 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 the L-Area. Wood storks have been observed in the Steel Creek delta, located about 9.8 kilometers (6 miles) south of the L-Area. The closest colony of red-cockaded woodpeckers to the 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 Creek floodplain about 12 kilometers (7.5 miles) northwest of L-Area (DOE 1995b). Consultation has been initiated 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 yet been evaluated. 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* (SHARP 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 the L-Area is limited. The area is in an archaeological site density zone that has the least potential for prehistoric sites of significance (DOE 1998f).

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 the L-Area before the development of the SRS in the early 1950s (DOE 1998f). Consultation has been initiated 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 initiated with the United Keetowah Band, Pee Dee Indian Association, and 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 geological 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 the F-Area and their occurrence in the 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 90.7 percent of all SRS employees reside (**Table 3–14**). In 1997, SRS employed 15,032 persons (5.8 percent of the regional economic area civilian labor force).

Table 3–14 Distribution of Employees by Place of Residence in the Savannah River Site Region of Influence, 1997

County	*Number of Employees*	*Total Site Employment (Percent)*
Aiken	6,981	53.9
Columbia	1,881	14.5
Richmond	1,755	13.5
Barnwell	932	7.2
Edgewell	210	1.6
Region of Influence Total	11,759	90.7

Source: DOE 1998b.

3.3.8.1 Regional Economy Characteristics

Between 1990 and 1996, the civilian labor force in the regional economic area increased 4.4 percent to the 1996 level of 259,174. In 1996, the unemployment rate in the regional economic area was 7.5 percent, which was greater than the unemployment rates for Georgia (4.6 percent) and South Carolina (6 percent that year).

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 1998b). 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 counties 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

3.3.8.3.1 Education

Community services include public education and public safety. In 1997, school districts providing public education in the region of influence were operating at capacities 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 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 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 the 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 42 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 the 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 3–15.** 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 3–15 Sources of Radiation Exposure to Individuals in the Savannah River Site Vicinity Unrelated to Savannah River Site 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 3–16.** 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.

Using a risk estimator of 500 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 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 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 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 fatal cancers estimated from SRS operations in 1997.

Table 3–16 Radiation Doses to the Public From Normal Savannah River Site Operations in 1997 (Total Effective Dose Equivalent)

Members of the Public	*Atmospheric Releases*		*Liquid Releases*		*Total*	
	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 (person-rem)[c]	None	5.5	None	2.4	100	7.9
Average individual within 80 kilometers (millirem)[d]	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, 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 conservatively assumed 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] 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.

SRS workers receive the same doses 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 3–17.** These doses fall within the radiological regulatory limits of 10 CFR 835. According to a risk estimator of 400 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 workforce of the more radiosensitive infant and child age groups.

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) are also presented in that report.

Table 3–17 Radiation Doses to Workers From Normal Savannah River Site 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. It has therefore established an administrative control level of 2,000 millirem per year; 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] 3,327 workers with measurable doses in 1997.

Sources: DOE 1995a, DOE 1998g.

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 the 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 1.1×10^{-3} picocuries per cubic meter in the F-Area, compared with the approximately 9.9×10^{-4} picocuries per cubic meter measured at the offsite control location. The concentration of plutonium-239 in the F-Area was 0 picocuries per cubic meter. 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 the L-Area. In 1997, the annual dose in the L-Area was 80 millirem (Arnett and Mamatey 1998b).

3.3.10.2 Chemical Environment

The background chemical environment important to human health consists of the atmosphere, which may contain hazardous chemicals that can be inhaled; drinking water, which may contain hazardous chemicals that can be ingested; and other environmental media through which people may come in contact with hazardous chemicals (e.g., surface water during swimming, soil through direct contact, or food). Hazardous chemicals can cause cancerous and noncancerous health effects.

Effective administrative and design controls that decrease hazardous chemical releases to the environment and help achieve compliance with permit requirements (e.g., air emissions and NPDES permit requirements) contribute to minimizing health impacts on the public. The effectiveness of these controls is verified through the use of monitoring information and inspection of mitigation measures. Health impacts on the public may occur via inhalation of air containing hazardous chemicals released to the atmosphere during normal SRS operations. Risks to public health from other possible pathways, such as ingestion of contaminated drinking water or direct exposure, are lower than those via the inhalation pathway.

The baseline concentrations are estimates of the highest existing offsite concentrations and represent the highest concentrations to which members of the public could be exposed. These concentrations are in compliance with applicable guidelines and regulations. Information on estimating the health impacts of hazardous chemicals is presented in Appendix E.

Exposure pathways to SRS workers during normal operation may include the inhalation of contaminants in the workplace atmosphere and direct contact with hazardous materials. The potential for health impacts varies among facilities and workers, and available information is insufficient for a meaningful estimate of impacts. However, workers are protected from workplace hazards through appropriate training, protective equipment, monitoring, substitution, and engineering and management controls. SRS workers are also protected by adherence to OSHA and EPA standards that limit workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. Appropriate monitoring that reflects the frequency and amounts of chemicals used in the operational processes ensures that these standards are not exceeded. Additionally, DOE requires that conditions 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.

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 waste; transuranic; mixed transuranic; low-level radioactive waste; mixed waste; hazardous; and nonhazardous. Waste generation rates and the inventory of stored waste from activities at SRS are provided in **Table 3–18. Table 3–19** summarizes the SRS waste management capabilities. More detailed descriptions of the waste management system capabilities at SRS are included in 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).

Table 3–18 Waste Generation Rates and Inventories at Savannah River Site

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 wastes.
[b] Generally, nonhazardous wastes are not held in long-term storage.
Source: DOE 1998b, except High-Level Radioactive Waste Generations Rate (DOE 1996c) and High-Level Radioactive Waste Inventory (DOE 1997a).

EPA placed SRS on the National Priorities List in December 1989. In accordance with CERCLA, DOE entered into a Federal Facilities Compliance Agreement with 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.

Table 3–19 Waste Management Capabilities at Savannah River Site

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	Online	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	Online				X	X	X	
F- and H-Area Effluent Treatment Facility	1,930,000	Online				X	X		
M-, L-, and H-Area Compactors	3,983	Online				X			
Non-Alpha Vitrification Facility	3,090	Planned				X	X	X	
M-Area Liquid Effluent Treatment Facility	999,000	Online					X		
M-Area Vendor Treatment Facility	2,470	Planned					X		
Savannah River Technology Center Ion Exchange Treatment Probe	11,200	Online					X		
E-Area Supercompactor	5,700	Planned				X			
Z-Area Saltstone Facility	28,400	Online					X		
Central Sanitary Wastewater Treatment Facility	1,030,000	Online							X
Storage Facility (cubic meters)									
Transuranic Storage Pads	34,400	Online		X	X				
Defense Waste Processing Facility Organic Waste Storage Tank	568	Online					X		
Liquid Waste Solvent Tanks	454	Planned					X		
M-Area Process Waste Interim Treatment/Storage Facility	8,300	Online					X		
Mixed Waste Storage Facilities (645-2N, -295, -43E)	1,905	Online					X		
Savannah River Technology Center Mixed Waste Storage Tanks	198	Online					X		
Long-Lived Waste Storage Building	1,064	Planned				X			
Solid Waste Storage Pads	2,657	Online					X	X	
Buildings 316-M, 710-B, 645-N, and 645-4N	2,515	Online					X	X	
M-Area Storage Pad	2,160	Online					X		
F- and H-Area Tank Farm	133,000	Online	X						
Defense Waste Processing Facility	2,286 canisters	Online	X						

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed	Haz	Non-Haz
Disposal Facility (cubic meters)									
Intermediate-Level Radioactive Waste Vaults	3,665	Online				X			
Low-Activity Waste Vaults	30,500	Online				X			
Low-Level Radioactive Waste Disposal Facility Slit Trenches	26,000	Planned				X			
Z-Area Saltstone Vaults	1,110,000	Online				X			

DWPF = Defense Waste Processing Facility, Haz = hazardous, HLW = high-level radioactive waste, LLW = low-level radioactive waste, TRU = transuranic
Sources: DOE 1998b, except High-Level Radioactive Waste (DOE 1996c).

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. These wastes are 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, flyash, 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 permanently immobilized 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 *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering and Environmental Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (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 1998b).

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 will 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 will 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 Creek. 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, wastes are 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 1998b).

Solid low-level radioactive waste is segregated into several categories to facilitate proper treatment, storage, and disposal. Solid low-level radioactive waste that radiates less than 200 millirem per hour at 5 centimeters (2 inches) from the unshielded container is considered low-activity waste. If it radiates 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 wastes are 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 1998b).

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 1998b).

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 per day (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 1998b). SRS-generated solid sanitary waste is sent to the Three Rivers Landfill (DOE 1998f) a permitted disposal facility. 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 wastes, 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 1998b). 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 1998d).

3.3.11.9 Preferred Alternatives From the Final Waste Management Programmatic Environmental Impact Statement and Associated Records of Decision

Preferred alternatives from the Waste Management Programmatic EIS (DOE 1997a) are shown in **Table 3–20** for the four waste types analyzed in this EIS. A decision on the future management of these wastes could result in the construction of new waste management facilities at SRS and the closure of other facilities. Decisions on the various waste types are expected to be announced in a series of Record of Decisions to be issued on the Waste Management Programmatic EIS. The transuranic waste Record of Decision was issued on January 20, 1998 (63 FR 3629), and the hazardous waste Record of Decision issued on August 5, 1998 (63 FR 41810). The transuranic waste Record of Decision states, ". . . .each of the Department's sites that currently has or will generate 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. More detailed information and DOE's alternatives for the future configuration of waste management facilities at SRS is presented in the Waste Management Programmatic EIS and the hazardous waste and transuranic waste Record of Decisions.

Table 3–20 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 prefers to treat SRS low-level radioactive waste on site. SRS could be selected as one of the regional disposal sites for low-level radioactive waste.[a]
Mixed	DOE prefers regionalized treatment of mixed waste at SRS. This includes the onsite treatment of SRS waste and could include treatment of some mixed waste generated at other sites. SRS could be selected as one of the regional disposal sites for mixed waste.[a]
Hazardous	DOE has decided to use commercial and onsite SRS facilities for treatment of SRS nonwastewater hazardous waste, and continue to use onsite facilities for wastewater hazardous waste.[c]

[a] From the Waste Management Programmatic EIS (DOE 1997a).
[b] From the Record of Decision for transuranic waste (63 FR 3629).
[c] From the Record of Decision for hazardous waste (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 the sodium-bonded spent nuclear fuel. It begins with a general discussion of the expected environmental impacts; the product and waste forms that would be generated from the proposed action; and the methodology for assessing health effects from radiological and chemical effluents. 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 the 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 IMPACTS

This *Draft 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 analyzed in detail involve construction other than internal building modifications for installing new equipment, the effects of construction 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, Analysis of Environmental Justice

4.1.1 Presentation of the Environmental Impacts

The primary impacts of concern are products and wastes, impacts on the public, and occupational health and safety associated with the various sodium-bonded spent nuclear fuel treatment processes. Additional impacts and topics covered in Chapter 4 include the following:

- Air Quality
- Water Quality
- Environmental Justice
- Socioeconomics
- Waste Management
- Transportation
- 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 will not occur, they will be extremely small, and/or they are covered by other analyses:

Land—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 are 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 limit the impacts to workers.

Noise—Noise impacts at the management sites should be minor and limited to noises 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 chronic radiation doses below 0.1 rad per day do not harm animal or plant populations (IAEA 1992). This is equivalent to 100 millirem per day for direct radiation and greater than 100 millirem per day for ingestion of plutonium. Compliance with DOE Order 5400.5 to limit the exposure of the most exposed member of the public to 100 millirem per year (i.e., about 0.3 millirem per day) makes it highly probable that dose rates to plants and animals in the same area would be less than 0.1 rad per day. Therefore, no radiological damage to plant and animal populations would be expected as the result of the sodium-bonded spent nuclear fuel treatment processes.

Chemicals emitted to the environment during routine processing activities from F-Canyon at SRS are presented in Section 4.5.1. In addition, Sections 4.5.4.1 and 4.5.4.2 contain modeled airborne concentrations for the chemicals emitted that have the potential to impact plants or animals. These chemicals would not impact plants or animals because either 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 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 environmental impact statement (EIS).

4.1.2 Products and Wastes

Generation—All the treatment processing alternatives in this EIS, except for direct disposal in high-integrity cans, would change sodium-bonded spent nuclear fuel into other forms. Driver and blanket sodium-bonded spent nuclear fuels are inputs—products and wastes are the outputs. The products and wastes are better suited for storage, transportation, and disposal or other disposition than the existing sodium-bonded fuel. The products and wastes fall into several distinct categories:

Materials to be managed as high-level radioactive waste would be generated at SRS and ANL-W. The final form would be solid ceramic, metal, or borosilicate glass inside stainless steel canisters. 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 concentrations (such as plutonium) 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 fuels would be made into low-enriched uranium ingots. The uranium products would be stored in secure facilities along with other uranium already in storage at ANL-W.

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 is currently stored at SRS). The 57 metric tons of depleted uranium are a small fraction of what is currently stored.

Separated plutonium resulting from PUREX processing of declad and cleaned blanket spent nuclear fuel at F-Canyon would be in a metal form. The separated plutonium would be stored in secure facilities along with the plutonium already in storage at SRS until decisions are made about its disposition. DOE would not use this plutonium for nuclear explosive purposes (DOE 1994a).

Low-level radioactive waste would be generated from all treatment technology alternatives 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.

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 1998f).

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 1998f).

4.1.3 General Radiological and Chemical Health Consequences

The methodologies used to evaluate potential radiological and chemical health effects 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 are very small fractions of these values, and total site doses will remain well within these DOE limits. For comparison, DOE estimates that the average individual in the United States receives a dose of approximately 350 millirem per year from all radiation sources combined, including natural and medical sources.

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 total population 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 environmental and occupational exposures to radiation. These effects include nonfatal cancers among the exposed population and genetic effects in subsequent generations. Nonfatal cancers and genetic effects are less probable consequences 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 dose-to-effect factors for fatal and nonfatal cancers are shown in **Table 4–1**.

Table 4–1 Risk of Latent Cancer Fatalities and Other Health Effects From Exposure to 1 Rem of Radiation[a]

Population [b]	*Latent Cancer Fatalities*	*Nonfatal Cancers*	*Genetic Effects*	*Total Detriment*
Workers	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 fatalities per rem of radiation dose. When applied to a population of individuals, units are 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.

The factors used in this EIS to relate a dose to its effect are 0.0004 latent cancer fatalities per person-rem for workers and 0.0005 latent cancer fatalities 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.

These concepts may be applied to estimate the effects of exposing a population to radiation. For example, if 100,000 people were each exposed only to natural background radiation (0.3 rem per year), 15 latent cancer fatalities per year would be expected (100,000 persons × 0.3 rem per year × 0.0005 latent cancer fatalities per person-rem = 15 latent cancer fatalities per year).

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 population 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 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, 1 latent cancer fatality would result; in exceptionally few groups, 2 or more latent cancer fatalities would occur. The average 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.

These same concepts apply to estimating the effects of radiation exposure on a single individual. Consider the effects, for example, of exposure to natural background radiation over a lifetime. The latent cancer fatality risk corresponding to a single individual's exposure to 0.3 rem per year over a (presumed) 72-year lifetime is:

> 1 person × 0.3 rem per year × 72 years × 0.0005 latent cancer fatalities per person-rem = 0.011 latent cancer fatalities, or slightly more than 1 chance in 100 of a latent cancer fatality.

Again, this is a statistical estimate. That is, the estimated effect of natural background radiation exposure on the exposed individual would produce a 1.1 percent chance that the individual would incur a latent cancer fatality. Presented another way, this method estimates that about 1 person in 91 would die of cancer induced by natural background radiation.

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. A recent examination of low radiation studies has reported that no statistically significant low-dose radiation study was found to support the linear no-threshold theory (Pollycove 1997). This finding is supported by the National Council of Radiation Protection and Measurements in a report on collective dose that states ". . . essentially no human data can be said to prove or even to provide direct support for the concept of collective dose with its implicit uncertainties of nonthreshold, linearity, and dose-rate independence with respect to risk" (NCRP 1995). Accordingly, calculations of health impacts based on the linear no-threshold theory may overstate the actual impacts of low radiation doses and should be viewed as an upper bound on the potential health effects.

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 the incident-free operation and accident conditions at management facilities. No hazardous chemicals are expected to be released from incident-free operation of the treatment technologies at ANL-W. The receptors considered in these evaluations include the offsite population in the vicinity of the sites and noninvolved workers located on site at SRS. Impacts also were evaluated for the maximally exposed offsite individual. The health effect endpoints evaluated in this analysis include excess latent cancers and chemical-specific noncancer health effects. The maximally exposed individual is located in the region with the highest estimated concentration. The hazardous chemical impacts are evaluated in terms of comparison to Emergency Response Planning Guideline. Emergency Response Planning Guidelines 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 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 interior of the fuel elements). Under this alternative, two options are evaluated:

a. The SBSNF EIS evaluates the impacts of the activities required to monitor and stabilize the sodium-bonded spent nuclear fuel as necessary for continued safe and secure storage indefinitely 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.)

b. The SBSNF EIS evaluates the impacts of direct disposal of sodium-bonded spent nuclear fuel in a geologic repository by packaging the fuel in high-integrity cans with minimal preparation.

Under both options, the EIS evaluates the impacts associated with activities required to clean and stabilize the waste materials generated during the Electrometallurgical Treatment Demonstration Project at ANL-W. Under this demonstration project, a total of approximately 1.6 metric tons of heavy metal of Experimental Breeder Reactor II (EBR-II) fuels consisting of about 1 metric ton of blanket spent nuclear fuel and 0.6 metric tons of driver spent nuclear fuel would be processed. The waste materials generated in this project currently are being transformed to ceramic and metallic waste forms, but the majority of wastes still would need stabilization. In addition, at the completion of the demonstration project, any remaining sodium-bonded spent nuclear fuel in the treatment facilities would be packaged and transferred to dry storage in the Radioactive Scrap and Waste Facility. Spent nuclear fuel transfer activities and waste processing activities would be completed in about two years after the necessary waste stabilization equipment is installed.

DOE is transferring all INEEL spent nuclear fuel, including the sodium-bonded spent nuclear fuel currently stored at the Idaho Nuclear Technology and Engineering Center (INTEC) Building 603 (wet storage), to dry storage. During transfer, each fuel can containing sodium-bonded fuel would be nondestructively examined to determine the fuel and can conditions and their suitability for storage. If any fuel can was found to be degraded and causing water inleakage, it would be repackaged and transferred to ANL-W for stabilization and/or recanning for storage. As stated in the amended Record of Decision for 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) (61 FR 9442), future sodium-bonded spent nuclear fuel transfers to Idaho would be packaged and stored at INTEC. If direct disposal of the sodium-bonded fuel becomes feasible, the stored fuel at the Radioactive Scrap and Waste Facility and at INTEC would be transferred to a dry spent nuclear fuel storage facility (to be built at INTEC) to be repackaged for offsite transport and disposal at a geologic repository.

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 indefinite storage. Both require that the 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 needs.

The impacts presented below would be applicable to either option considered under the No Action Alternative. The only impact that is different between the two options is the volume of high-level radioactive waste presented in Table 4-8. All other impacts are identical for each of the two options.

4.2.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that activities under either option of this alternative would have a small impact on existing air quality at ANL-W, as any nonradiological emissions would be very low and well below the regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

Radiological Gaseous Emissions

Potential radiological releases from spent nuclear fuels during storage periods at INEEL were estimated based on the information provided in the No Action Alternative in the Programmatic Spent Nuclear Fuel EIS (DOE 1995a). Normal spent nuclear fuel storage activities would produce radiological air emissions that are small compared to radiological air emissions from other activities at INEEL, such as calciner operations at INTEC and reactor operations at the Test Reactor Area and ANL-W. The current estimates of radiological emissions are significantly lower than those used for the evaluation of impacts in the Programmatic Spent Nuclear Fuel EIS. For example, in 1997, the ANL-W facilities released only 1.14 curies of krypton-85 (DOE 1998c) as compared to the estimate of 13,000 curies used earlier in the Programmatic Spent Nuclear Fuel EIS. However, degradation of sodium-bonded spent nuclear fuel during storage cannot be ruled out. It is expected that a percentage of fuel would be degraded during storage, allowing the gaseous fission products to enter the storage environment. During fuel handling for examination and repackaging, these fission gases would be released to the environment. Since the extent of fuel degradation would not be known in advance, for the purposes of this EIS the estimates of air emissions during handling operations are conservatively based on the radiological gaseous emissions provided in the Programmatic Spent Nuclear Fuel EIS and adjusted for the percentage of sodium-bonded fuel to that of the total spent nuclear fuel inventory at INEEL. Therefore, annual radiological gaseous emissions are estimated to be between 0 and 460 curies of tritium/carbon-14 and between 0 and 7,120 curies of krypton-85 (DOE 1995a). These estimates of air emissions are conservative and would bound any potential releases that may occur during handling operation, (these releases correspond to an inventory of about 10 percent of degraded driver spent nuclear fuels). The handling operation for repackaging is estimated to last about two years.

4.2.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River are not 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. Current operating and monitoring practices would continue for National Pollutant Discharge Elimination System (NPDES) stormwater and liquid effluent discharges associated with facilities at ANL-W.

Radiological Liquid Effluent

No radiological liquid effluents 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 if the number of workers at ANL-W were to decrease. The current water use at ANL-W is 188 million liters (49.6 million gallons per year) per year .

Nonradiological Liquid Effluent

For either option of this alternative, no nonradiological liquid effluents or wastes would be discharged to groundwater.

Radiological Liquid Effluent

For either option of this alternative, no radiological liquid effluents 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 623 indirect jobs. The total potential loss of 973 represents a 1 percent decrease in the regional economic area civilian labor force, which was estimated to be 150,835 in 1996 (DOE 1998d).

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 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.

Both options of the No Action Alternative would therefore most likely not result in any noticeable change in the existing regional economy, population and housing characteristics, and community services within the region of influence at ANL-W (see Section 3.2.8). Overall expenditures and employment at INEEL should remain relatively constant through 2010, which would, in turn, tend to maintain economic and demographic characteristics within the region of influence.

4.2.4 Public and Occupational Health and Safety

The assessments of potential radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in **Tables 4–2** and **4–3** for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in **Tables 4–4** and **4–5**. The impacts from hazardous chemical releases during accident conditions are presented in **Table 4–6**. 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 this alternative, radioactive releases from normal operations associated with spent nuclear fuel storage activities at ANL-W and INTEC would be small. Annual radiation doses to the public from these activities were calculated for the No Action Alternative in the Programmatic Spent Nuclear Fuel EIS based on a total INEEL spent nuclear fuel inventory of approximately 274 metric tons of heavy metal. The inventory of sodium-bonded spent nuclear fuel in storage at ANL-W and INTEC represents about 60 metric tons of heavy metal, or about 22 percent of the INEEL inventory identified in the Programmatic Spent Nuclear Fuel EIS. For this SBSNF EIS, radiological impacts from normal operations associated with storage of sodium-bonded spent nuclear fuel are estimated to be about 22 percent of the impacts calculated for the No Action Alternative in the Programmatic Spent Nuclear Fuel EIS.

Calculated maximum annual radiological impacts to the public are given in Table 4–2. The impacts are calculated for two receptor groups: the general public living within 80 kilometers (50 miles) of INEEL 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.

Table 4–2 Annual Radiological Impacts to the Public From Operational Activities Associated With the No Action Alternative

Receptor	*No Action Alternative*
Population Dose Within 80 Kilometers (50 Miles)	
Dose (person-rem)[a]	0.022
Latent cancer fatalities	0.000011
Annual Dose to the Maximally Exposed Offsite Individual	
Dose (millirem)[a]	0.00077
Latent cancer fatality risk	3.9×10^{-10}
Percent of natural background[b]	0.00021
Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)[c]	
Dose (millirem)	0.000092
Latent cancer fatality risk	4.6×10^{-11}

[a] Based on 22 percent of the dose reported in Volume 1, Appendix B, Programmatic Spent Nuclear Fuel EIS (DOE 1995a).

[b] The annual natural background radiation level at INEEL is 360 millirem for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,520 person-rem. The site population in 2010 was assumed to be representative of the population over the operational period evaluated.

[c] 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).

Occupational doses were also estimated based on worker doses calculated for the No Action Alternative in the Programmatic Spent Nuclear Fuel EIS. 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; see Table 4–3 (see Section 4.3.4).

Table 4–3 Annual Radiological Impacts to Workers From Operational Activities Associated With the No Action Alternative

Impact	*No Action Alternative*
Worker[a]	
Average worker dose (millirem per year)	60
Latent cancer fatality risk	0.000024
Total dose (person-rem per year)	22
Latent cancer fatalities	0.0088

[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 N441.1.

Source: ANL 1999.

As shown in Tables 4-2 and 4-3:

- The annual dose to the maximally exposed offsite individual would be 0.00077 millirem per year, with an associated 3.9×10^{-10} risk per year of developing a fatal cancer (or one in 2.5 billion years).

- The collective dose to the population within 80 kilometers (50 miles) of the storage facilities at INEEL would be 0.022 person-rem per year, with an associated 0.000011 latent cancer fatalities per year (or one in 90,900 years).

- 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 in 113 years).

Hazardous Chemical Impacts

It is expected that the hazardous chemical impacts associated with either option of this alternative at ANL-W would be negligible, because any emissions of hazardous chemicals from activities under this alternative would be very low. The existing chemical environment is presented in Section 3.2.10.2.

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 residual wastes generated during the electrometallurgic treatment demonstration project at ANL-W have the potential to involve accident scenarios similar to those evaluated for Alternative 1. However, the consequences associated with these accident scenarios are lower because of the limited quantities of residual wastes to be stabilized. Accidents associated with spent nuclear fuel transfer activities also could occur during the time when spent nuclear fuel is removed from the Radioactive Scrap and Waste Facility to prepare it for packaging and offsite shipment to a repository. It is estimated that spent nuclear fuel transfer and the waste stabilization activities would occur over a two-year time period.

During the time that sodium-bonded spent nuclear fuel is in dry storage at the Radioactive Scrap and Waste Facility, it is in a very safe and stable configuration, and no reasonably foreseeable accident scenarios could be identified. Sodium-bonded spent nuclear fuel currently in wet storage at INTEC would be transferred to dry storage facilities at INTEC. Handling accidents could occur during transfer activities at INTEC, similar to the accident scenario evaluated for ANL-W. However, because INTEC is further away from the INEEL site boundary and major population centers compared to ANL-W, the accident impacts at INTEC would be less than those for ANL-W.

Table 4–4 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 95^{th} percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50^{th} percentile meteorological conditions. The 50^{th} percentile condition represents the median meteorological condition, and is defined as that for which more severe conditions occur 50 percent of the time. The 95^{th} 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 two reasons: first, no adequate method exists for calculating meaningful consequences at or near the location where the accident occurs. Second, 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), the evaluations required by such a system, and the products derived from such evaluations

(e.g., procedures, programs, emergency plans). 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 presents very severe nonradiological effects to the involved workers. In such a scenario, the workers are likely to be hurt or 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 for the same receptors are summarized in Table 4–5.

Table 4–4 Accident Frequency and Consequences for 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* [b]	*Dose (person-rem)*	*Latent Cancer Fatalities* [c]	*Dose (millirem)*	*Latent Cancer Fatality* [b]
Salt powder spill in the Hot Fuel Examination Facility cell	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	3.5×10^{-6}	0.22	8.7×10^{-8}
Design-basis seismic event	0.008	12	6.0×10^{-6}	1.4	0.00070	4.7	1.9×10^{-6}
Salt spill during transfer	1×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Beyond-design-basis seismic event	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.

Table 4–5 Annual Cancer Risks Due to Accidents for 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}	3.5×10^{-9}	8.7×10^{-11}
Design-basis seismic event	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Salt spill during transfer	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Beyond-design-basis seismic event	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 the 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 in 20.8 million years) and 1.5×10^{-8} per year (or one in 66.7 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 5.6×10^{-6} per year (or one in 178,600 years).

Hazardous Chemical Impacts

Nonradiological hazardous chemical impacts are evaluated in terms of comparison to Emergency Response Planning Guidelines. Emergency Response Planning Guideline 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 4–6.

Table 4–6 Hazardous Chemical Accident Impacts for the No Action Alternative

Accident	*Frequency (event per year)*	*Receptor Location*	*Exposure*
Uranium-handling accident	0.01	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker at 100 meters	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 Section 4.2.4, operations conducted under either option of this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed individual over the 35 years' duration of interim storage operation and removal from the INEEL site (which is assumed to occur by 2035) would be 0.000014 (or 1 chance in 71,400), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.00039 (or 1 chance in 2,560). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small regardless of the racial and ethnic composition of the population and independent of the economic status of individuals comprising the population. Operation of spent nuclear fuel storage facilities at ANL-W and INTEC would have no disproportionately high and adverse effects on minority or low-income populations.

4.2.6 Waste Management

Various types of waste would be generated associated with sodium-bonded spent nuclear fuel storage activities at ANL-W and INTEC, including transuranic waste, mixed transuranic waste, low-level radioactive waste, mixed waste, hazardous, and nonhazardous wastes. 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 Demonstration Project cladding hull waste and electrorefiner salt are stabilized to metallic and ceramic high-level radioactive waste forms for ultimate disposal. **Table 4–7** shows the

anticipated categorization of these waste types and their expected interim storage and final disposal locations. The quantities of ceramic and metal waste forms generated, along with other generated wastes, are presented in **Table 4–8**.

Table 4–7 Summary of Process Waste Material Categories for the No Action Alternative

Waste Stream	*Category*	*Interim Storage Location*	*Final Disposal Location*
Process Wastes			
Fuel hardware	Low-level radioactive waste	None	Radioactive Waste Management Complex\
Metal waste form	High-level radioactive waste	Radioactive Scrap and Waste Facility	Offsite (proposed geologic) repository
Ceramic waste form	High-level radioactive waste	Radioactive Scrap and Waste Facility	Offsite (proposed geologic) repository
Other Associated Process Wastes			
less than 100 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 Wastes			
Electrorefiner cadmium	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant
Equipment less than 100 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). Wastes in this category may be treated by the Advanced Mixed Waste Treatment Project and then disposed of at the Waste Isolation Pilot Plant."

Direct Process Wastes

Under either option of the No Action Alternative, small amounts of metal and ceramic high-level radioactive waste would be produced at ANL-W as a result of the completion of the Electrometallurgical Treatment 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 metal waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. Both the ceramic and metal waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–8 are for the standardized canisters required for disposal of these materials.

Table 4–8 Amount of Wastes Generated for the No Action Alternative[a]

Waste Stream	Total Waste Generated	
	Volume (Cubic Meters)	Mass (Kilograms)
Direct Process Wastes		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	9.4	14,000
High-level radioactive metal waste	0.6	460
Spent nuclear fuel	92/142[b]	72,000
Other Associated Process Wastes		
Low-level radioactive wastes	700	142,000
Transuranic wastes	8.4	3,000
Mixed wastes	35	19,000
Sanitary wastes	2,500	867,000
Deactivation Wastes		
Low-level radioactive wastes	112	38,000
Transuranic wastes	1.6	853
Mixed wastes	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] Volumes for interim storage/direct disposal.

Source: ANL 1999.

The metal and ceramic high-level radioactive waste generated as a result of the electrometallurgical treatment demonstration at ANL-W would be stored temporarily for 10 to 15 years 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 steel storage liners in which the waste would be stored, and by the soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste cans containing the metal 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 Wastes

Other associated process low-level radioactive wastes would be generated as a result of the deactivation and 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 unforseen problems while storing the fuel in the Radioactive Scrap and Waste Facility. These wastes are the result of 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 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, and most years would result in approximately 17 cubic meters (600 cubic feet). 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, and the total of 700 cubic meters (24,700 cubic feet) represents approximately 0.6 percent of the total Radioactive Waste Management Complex disposal inventory.

Other Associated Process Transuranic Wastes

Other associated process transuranic wastes would be generated at ANL-W under either option of the No Action Alternative from decontamination activities, repair and maintenance of items, and miscellaneous work associated with demonstration fuel processing or other activities. Transuranic wastes 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 per year (35 cubic feet per year), and most years would result in approximately 0.2 cubic meters (7 cubic feet). 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 the No Action Alternative is approximately 8.4 cubic meters (300 cubic feet), which is 0.005 percent of the estimated total volume of transuranic waste to be emplaced at the Waste Isolation Pilot Plant.

Other Associated Process Sanitary Wastes

These sanitary wastes that are nonradioactive and nonhazardous would continue to be generated under either option of the No Action Alternative. These solid wastes 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 solid sanitary waste generated and disposed of under this alternative is approximately 0.1 percent of the INEEL landfill volume.

Other Associated Process Mixed Wastes

These 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 1996b).

Deactivation Wastes

A variety of wastes would be generated as part of deactivation activities at ANL-W. These would include process equipment and process material such as electrorefiner cadmium. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes 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 wastes 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 electrorefiner 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. Under the No Action Alternative, it is anticipated that the deactivation waste volumes would be generated over a period of one year. The total deactivation wastes represent approximately 17 percent over the total incidental waste (excluding sanitary wastes) requiring disposal.

Once the residual high-level radioactive wastes from the Electrometallurgical Treatment Demonstration Project have been stabilized, which is estimated to require about two years, the types and quantities of wastes generated under the No Action Alternative would be consistent with current activities at INEEL, as presented in Section 3.2.11.1.

4.3 ALTERNATIVE 1: ELECTROMETALLURGICAL TREATMENT OF 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 a low-enriched uranium ingot, 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, metal 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 including plutonium elements, and a metal waste form consisting of noble metal fission products and cladding hulls from the spent nuclear fuel. The low-enriched uranium metal ingot would be stored at the Zero Power Physics Reactor Material Storage Building. The ceramic and metal 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.

4.3.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that this alternative at ANL-W will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

Radiological Gaseous Emissions

Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that are 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. 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 occurs during the first five years of the electrometallurgical treatment process, where a combination of EBR-II blanket and driver spent nuclear fuel elements are processed. During these five years, about 600 kilograms of heavy metal driver spent nuclear fuel and about 4,400 kilograms of heavy metal 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; see Appendix E.4.1 for details. The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.3.4.

4.3.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River are not 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 NPDES stormwater and liquid effluent discharges associated with facilities at ANL-W.

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 1996a).

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 4–9** and **4–10** for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in **Tables 4–11** and **4–12**. The impacts from hazardous chemical releases during accident conditions are presented in **Table 4–13**. 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 six years, after which only Fermi-1 blanket spent nuclear fuel 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 radiological impacts to the public are given in Table 4–9. The impacts are calculated for two receptor groups: 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 gases (about 1 percent of which conservatively was 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.

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 total worker population 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, the cumulative worker population dose would be 286 person-rem, leading to a risk of 0.11 latent cancer fatalities (see Table 4–10).

Table 4–9 Annual Radiological Impacts to the Public From Operational Activities Associated With Alternative 1

Receptor	*Alternative 1*		
	Electrometallurgical Treatment of Driver Spent Nuclear Fuel	*Electrometallurgical Treatment of Blanket Spent Nuclear Fuel*	*Total*
Population Dose Within 80 Kilometers (50 Miles) in the Year 2010			
Dose (person-rem)	0.0028	0.000084	0.0029
Latent cancer fatalities	1.4×10^{-6}	4.2×10^{-8}	1.5×10^{-6}
Annual Dose to the Maximally Exposed Offsite Individual			
Dose (millirem)	0.00033	0.00001	0.00034
Latent cancer fatality risk	1.6×10^{-10}	5.0×10^{-12}	1.7×10^{-10}
Percent of natural background[a]	0.000092	2.7×10^{-6}	0.000094
Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)[b]			
Dose (millirem)	0.000012	3.5×10^{-7}	0.000012
Latent cancer fatality risk	6.0×10^{-12}	1.8×10^{-13}	6.0×10^{-12}

[a] The annual natural background radiation level at INEEL is about 360 millirem for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.

[b] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of the ANL-W in the year 2010 (240,338).

Table 4–10 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 1

Impact	*Alternative 1*
Worker[a]	
Total dose (person-rem per year)	22
13-year fatal cancer risk	0.11
Average worker dose (millirem per year)	60
13-year fatal cancer risk	0.00031

[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 N441.1.

Source: ANL 1999.

As shown in Tables 4–9 and 4–10:

- The annual dose to the maximally exposed offsite individual would be 0.00034 millirem per year, with an associated 1.7×10^{-10} risk per year of developing a fatal cancer (or one in 5.9 billion years).
- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0029 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one in 667,000 years).
- 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 in 113 years).

Hazardous Chemical Impacts

It is expected that hazardous chemical impacts associated with this alternative will be negligible, as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described in Section 3.2.10.2.

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 are 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 4–11 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 for the same receptors are summarized in Table 4–12.

Table 4–11 Accident Frequency and Consequences for 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)*	*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}
Earthquake (design-basis earthquake)	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Earthquake (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	0.00012	6.2×10^{-11}	0.000027	1.3×10^{-8}	1.1×10^{-6}	4.4×10^{-13}
Salt transfer drop	1×10^{-7}	0.052	2.6×10^{-8}	0.0062	3.1×10^{-6}	0.17×10^{-3}	6.8×10^{-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}

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)	Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Earthquake (design-basis earthquake at Hot Fuel Examination Facility)	0.008	3.3	1.6×10^{-6}	0.4	0.0002	11	4.4×10^{-6}
Earthquake (beyond-design-basis earthquake)	0.00001	0.071	0.00035	83	0.041	38	0.000019

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Table 4–12 Annual Cancer Risks Due to Accidents for 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}	1.8×10^{-8}	3.4×10^{-12}
Earthquake (design-basis earthquake)	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Earthquake (beyond-design-basis earthquake)	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Salt powder spill	6.2×10^{-13}	1.3×10^{-10}	4.4×10^{-15}
Salt transfer drop	2.6×10^{-15}	3.1×10^{-13}	6.8×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.5×10^{-9}	8.7×10^{-11}
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Earthquake (design-basis earthquake)	1.3×10^{-8}	1.6×10^{-6}	3.5×10^{-8}
Earthquake (beyond-design-basis earthquake)	3.5×10^{-9}	4.1×10^{-7}	1.7×10^{-9}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For the 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 in 4.5 million years) and 3.5×10^{-8} per year (or one in 28.6 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one in 76,920 years).

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to Emergency Response Planning Guidelines. Emergency Response Planning Guideline 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 4–13.

Table 4–13 Hazardous Chemical Accident Impacts for Alternative 1

Accident	*Frequency (event per year)*	*Receptor Location*	*Exposure*
Uranium-handling accident	0.01	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker at 100 meters	Cadmium: less 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 Section 4.3.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the 13 years of electrometallurgical treatment operation would be 2.2×10^{-9} (or one chance in 454 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000020 (or one chance in 50,000). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small regardless of the racial and ethnic composition of the population, and independent of the economic status of individuals comprising the population. Operation of electrometallurgical processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

4.3.6 Waste Management

Electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W would generate process wastes from treatment operations, other associated process wastes from normal support operations and deactivation wastes following the conclusion of operations. Process wastes would include fuel hardware and metal and ceramic high-level radioactive wastes. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes 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 4–7 (see Section 4.2.6).

Estimates of the total amount of other associated process waste generated as a result of electrometallurgical treatment at ANL-W are provided in **Table 4–14**. These values 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 4–14 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.

Table 4–14 Amounts of Wastes Generated for Alternative 1[a]

Waste Stream	*Total Waste Generated*	
	Volume (Cubic Meters)	*Mass (Kilograms)*
Direct Process Wastes		
Fuel hardware (low-level radioactive waste)	12	6,600
High-level radioactive ceramic waste	78	120,000
High-level radioactive metal waste	6.3	9,000
Other Associated Process Wastes		
Low-level radioactive wastes[b]	706	143,000
Transuranic wastes	12.5	5,400
Mixed wastes	35.3	19,000
Sanitary wastes	4,960	1.72×10^{-6}
Deactivation Wastes		
Low-level radioactive wastes[b]	143	48,000
Transuranic wastes	1.6	853
Mixed wastes	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 are accomplished during this time period.

[b] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Direct Process Wastes

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, metal 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 metal 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 metal waste form. Both the ceramic and metal waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–14 are for the standardized canisters required for disposal of these materials.

The metal and ceramic high-level radioactive waste generated would be temporarily stored for 10 to 15 years 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, and shielding would be provided by a combination of steel storage liners storing the waste, and the shielding provided by soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste

cans containing the metal 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 Wastes

These low-level radioactive wastes 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 will 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 Wastes

These transuranic wastes would be generated by decontamination activities, repair and maintenance of items, and miscellaneous work associated with the electrometallurgical processing. Transuranic wastes 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 approximately 13 cubic meters (459 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 Wastes

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 1996b).

Deactivation Wastes

A variety of wastes would be generated as part of deactivation activities associated with electrometallurgical treatment processing at ANL-W. These would include process equipment and process material, such as electrorefiner cadmium. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes 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 wastes 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. If the deactivation waste volume is generated in a single year, the wastes would represent an increase of approximately 3.5 times the annual waste generated by electrometallurgical treatment requiring disposal. The total deactivation wastes represent approximately 30 percent over the total incidental waste (excluding sanitary wastes) requiring disposal.

4.4 ALTERNATIVE 2: PACKAGE BLANKET FUEL IN HIGH-INTEGRITY CANS AND TREAT (ELECTROMETALLURGICAL TREATMENT) 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 (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 fuels 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 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 be completed by 2009.

4.4.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that this alternative at ANL will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

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. 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 occurs 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 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. 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 are not 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.

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 1996a).

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 4–15** and **4–16** for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in **Tables 4–17** and **4–18**. The impacts from hazardous chemical releases during accident conditions are presented in **Table 4–19**. 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 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 cleaning of blanket spent nuclear fuels and treatment of driver spent nuclear fuels 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 radiological impacts to the public are given in Table 4–15. The impacts are calculated for two receptor groups: 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 gases (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.

Table 4–15 Annual Radiological Impacts to the Public From Operational Activities Associated With Alternative 2

Receptor	*Alternative 2*		
	Electrometallurgical Treatment of Driver Spent Nuclear Fuel	*Clean and Place Blanket Spent Nuclear Fuel in High-Integrity Cans*	*Total*
Population Dose Within 80 Kilometers (50 Miles) in the Year 2010			
Dose (person-rem)	0.0028	0.00028	0.0031
Latent cancer fatalities	1.4×10^{-6}	1.4×10^{-7}	1.6×10^{-6}
Annual Dose to the Maximally Exposed Offsite Individual			
Dose (millirem)	0.00033	0.000048	0.00038
Latent cancer fatality risk	1.6×10^{-10}	2.4×10^{-11}	1.9×10^{-10}
Percent of natural background[a]	0.000092	0.000013	0.00011
Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)[b]			
Dose (millirem)	0.000012	1.2×10^{-6}	0.000013
Latent cancer fatality risk	6.0×10^{-12}	6.0×10^{-13}	6.6×10^{-12}

[a] The annual natural background radiation level at INEEL is 360 millirem for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.

[b] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of the ANL-W in the year 2010 (240,338).

Table 4–16 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 total worker population 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, the cumulative worker population dose would be 198 person-rem, leading to a risk of 0.079 latent cancer fatalities.

Table 4–16 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 2

Impact	*Alternative 2*
Worker[a]	
Total dose (person-rem per year)	22
Nine-year fatal cancer risk	0.079
Average worker dose (millirem per year)	60
Nine-year fatal cancer risk	0.00022

[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 N441.1.

Source: ANL 1999.

As shown in Tables 4–15 and 4–16:

- 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 fatal cancer (or one in 5.3 billion years).
- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0031 person-rem per year, with an associated 1.6×10^{-6} latent cancer fatalities per year (or one in 625,000 years).
- 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 in 113 years).

Hazardous Chemical Impacts

It is expected that hazardous chemical impacts associated with this alternative at ANL-W will be small, as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described in Section 3.2.12.2.

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 fuels 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 is 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 driver spent nuclear fuel processing takes 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 is 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, the higher frequency design-basis seismic event was analyzed for blanket spent nuclear fuels only. Table 4–17 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 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 4–18.

Table 4–17 Accident Frequency and Consequences at ANL-W for 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)*	*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}
Earthquake (design-basis earthquake at Hot Fuel Examination Facility)	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Earthquake (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}	0.689	0.00034	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 4–18 Annual Cancer Risks Due to Accidents at ANL-W for 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}	1.8×10^{-8}	3.4×10^{-12}
Earthquake (design-basis earthquake)	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Earthquake (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}	3.5×10^{-9}	8.7×10^{-11}
Sodium fire	2.4×10^{-8}	2.7×10^{-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 in 4.5 million years) and 1.5×10^{-8} per year (or one in 66.7 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one in 76,920 years).

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 fuels using the electrometallurgical treatment process are summarized in Table 4–19.

Table 4–19 Hazardous Chemical Impacts Due to Accidents at ANL-W for Alternative 2

Accident	*Frequency (event per year)*	*Receptor Location*	*Exposure*
Uranium-handling accident	0.01	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.008	Noninvolved worker at 100 meters	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker at 100 meters	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 Section 4.4.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the nine-year period of cleaning blanket spent nuclear fuel for placement in high-integrity cans and treatment of driver spent nuclear fuels using the electrometallurgical treatment would be 1.7×10^{-9} (or one chance in 588 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000014 (or one chance in 71,400). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small, regardless of the racial and ethnic composition of the population, and independent of the economic status of individuals comprising the population. Operation of high-integrity can and electrometallurgical processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

4.4.6 Waste Management

This alternative would generate process wastes from treatment operations, incidental wastes from normal support operations, and deactivation wastes following the conclusion of operations. Process wastes would include fuel assembly hardware and metal and ceramic high-level radioactive wastes. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes 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 4–7 (see Section 4.2.6).

Estimates of the total amount of other associated process wastes generated as a result of Alternative 2 are provided in **Table 4–20**. These values 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 4–20 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 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.

Direct Process Wastes

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, metal 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 metal 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 metal waste form. Both the ceramic and metal waste would be categorized as high-level radioactive waste.

Table 4–20 Amounts of Wastes Generated for Alternative 2[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Wastes		
Fuel assembly hardware (low-level radioactive waste)	12.5	6,000
High-level radioactive ceramic waste	16.3	24,400
High-level radioactive metal waste	2.0	2,500
Spent nuclear fuel	25.2	63,000
Other Associated Process Wastes		
High-level radioactive wastes	0.4	220
Low-level radioactive wastes[b]	555	113,000
Transuranic wastes	9.1	3,800
Mixed wastes	27.5	14,800
Sanitary wastes	4,960	1.72×10^6
Deactivation Wastes		
Low-level radioactive wastes[b]	166.2	56,000
Transuranic wastes	1.6	853
Mixed wastes	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 are accomplished during this time period.

[b] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

The packaged spent nuclear fuel volume is based on placing the blanket spent nuclear fuel in high-integrity cans which will be placed in standardized canisters. The volumes of waste forms provided in Table 4–20 are for the standardized canisters required for disposal of these materials.

The metal and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel and packaged spent nuclear fuel would be temporarily stored for 10 to 15 years 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 and shielding would be provided by a combination of steel storage liners storing the waste, and shielding provided by the soil surrounding the liners. When an offsite (proposed 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

These high-level radioactive wastes 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 absorbant 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 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 Wastes

These low-level radioactive wastes 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 will 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 Wastes

These transuranic wastes would be generated by Alternative 2 from decontamination activities, repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic wastes 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 approximately 9 cubic meters (318 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 Wastes

These mixed wastes 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 1996b).

Deactivation Wastes

A variety of wastes would be generated as part of deactivation activities associated with electrometallurgical treatment processing at ANL-W. These would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes 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 wastes 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. Deactivation waste volume is generated in two years. The total deactivation wastes represent an additional 30 percent over the total incidental waste (excluding sanitary waste) requiring disposal.

4.5 ALTERNATIVE 3: DECLAD AND CLEAN BLANKET FUEL AND TREAT (ELECTROMETALLURGICAL TREATMENT) 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 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 fuels 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 be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its shipment to SRS could start in 2003 and be completed 2009. PUREX processing of blanket spent nuclear fuel at SRS could be completed by 2010.

4.5.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that this alternative at ANL will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). 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 4–21** along with the total estimated site air pollutant concentrations. The concentrations for the alternative are based on information in the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement* (SRS Spent Nuclear Fuel Management Draft EIS) (DOE 1998g). The concentrations have been adjusted to account for the increased mass of sodium-bonded spent nuclear fuel. The total concentrations are equal to the concentrations for the alternative, plus the baseline concentrations from Section 3.3.3.1. The concentrations are compared to their corresponding ambient air quality standards. Only those air pollutants that are expected to be emitted under this alternative and that 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 4–21 Nonradiological Air Quality Concentrations Associated With Alternative 3 at SRS for Comparison With Ambient Air Quality Standards

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*	*Alternative 3 Concentration (micrograms per cubic meter)*	*Total Site Concentration (micrograms per cubic meter)*
Criteria pollutants				
Carbon monoxide	8 hours 1 hour	10,000 40,000	1.22 9.06	633.02 5023.66
Nitrogen dioxide	Annual	100	3.11	11.91
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 150 150	less than 0.01 0.11 NA	4.8 80.72 NA
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 65	NA NA	NA NA
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	less than 0.01 0.12 0.91	16.3 215.52 691.1 1
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	0.12 0.14 0.66 2.4
Total suspended particulates	Annual	75	less than 0.01	43.3
Hazardous and other toxic compounds				
1,1,1-trichloroethane	24 hours	9,550	less than 0.01	less than 22.01
Benzene	24 hours	150	0.01	31.01
Ethanolamine	24 hours	200	less than 0.01	less than 0.02
Ethyl benzene	24 hours	4,350	less than 0.01	less than 0.13
Ethylene glycol	24 hours	650	less than 0.01	less than 0.09
Formaldehyde	24 hours	15	less than 0.01	less than 0.02
Glycol ethers	24 hours	No standard	less than 0.01	less than 0.02
Hexachloronaphthalene	24 hours	1	less than 0.01	less than 0.02
Hexane	24 hours	900	0.01	0.08
Manganese	24 hours	25	less than 0.01	less than 0.11
Methyl alcohol	24 hours	1,310	less than 0.01	less than 0.52
Methyl ethyl ketone	24 hours	14,750	less than 0.01	less than 1
Methyl isobutyl ketone	24 hours	2,050	less than 0.01	less than 0.52
Methylene chloride	24 hours	8,750	0.01	1.81
Naphthalene	24 hours	1,250	less than 0.01	less than 0.02
Nitric acid	24 hours	125	0.28	6.98
Phenol	24 hours	190	less than 0.01	less than 0.04
Phosphorous	24 hours	0.5	less than 0.01	less than 0.01
Sodium hydroxide	24 hours	50	less than 0.01	less than 0.02
Toluene	24 hours	2,000	0.01	1.61
Trichloroethane	24 hours	6,750	less than 0.01	less than 1.01

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)	Alternative 3 Concentration (micrograms per cubic meter)	Total Site Concentration (micrograms per cubic meter)
Vinyl acetate	24 hours	176	less than 0.01	less than 0.03
Xylene	24 hours	4,350	0.02	3.82

Source: Bickford et. al. 1997, plus baseline concentrations from Section 3.3.3.1.
NA = Not Available.

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 occurs 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. 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.

Since declad and clean fuels are packaged and sent to SRS, some gaseous fission products are expected to be present in that fuel. However, it was conservatively assumed that all gaseous fission products in the blanket spent nuclear fuels 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 fuels and treatment of driver spent nuclear fuels 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 fuels 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 3-7.

Nonradiological Liquid Effluent

The major sources of liquid effluents 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 effluents from this processing. Liquid effluents associated with PUREX processes would use these facilities and the existing permitted outfalls (Section 3.3.4.1). Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility and discharged through an existing NPDES-permitted outfall (G-10). Since employment would not increase as a result of processing these fuels, 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 1998g).

Process cooling water treatment would result in releases to Upper Three Runs Creek from the F-Area, as shown in **Table 4–22**.

Table 4–22 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	NR	(d)
Chromium	0.02	ND	ND	0.1
Copper	0.01	0.018	0.015	1
Manganese	0.01	0.039	0.052	0.05
Nickel	0.05	ND	ND	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.
Key: ND = not detected; NR = not reported.
Sources: Arnett and Mamatey 1998, DOE 1998g.

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 Quality 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 effluents from processing declad and cleaned blanket spent nuclear fuels at F-Canyon were 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 6-month operation of F-Canyon. **Table 4–23** provides a list of potential radiological isotopes that could be released to the surface water during processing of about 57 metric tons of heavy metal of blanket spent nuclear fuel (see Appendix E, Sections E.3, for details).

Table 4–23 Estimated Radiological Liquid Effluents During PUREX Processing of Blanket Spent Nuclear Fuels

Isotope	*Curies Released*
Tritium	1.54
Strontium-89/Strontium-90	0.000031
Ruthenium-103/Ruthenium-106	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.

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 (1.3 billion gallons) per year of groundwater (DOE 1998g).

Nonradiological Liquid Effluent

No nonradiological chemicals would be discharged to groundwater from PUREX processing of blanket spent nuclear fuels 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 fuels 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 4–24** and **4–25** for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in **Tables 4–28** and **4–29**. The impacts from hazardous chemical releases during accident conditions are presented in **Table 4–32**. 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 fuels being processed at SRS are 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 are still 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 radiological impacts to the public are given in Table 4–24. The impacts are calculated for two receptor groups: 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 are from tritium gases (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 Draft EIS (DOE 1998g) (see Appendix E, Section E.4.3, for details). The doses and duration from decladding and cleaning blanket spent nuclear fuels and treatment of driver spent nuclear fuels 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 4–25.

Table 4–25 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 cumulative 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 fuels at ANL-W is 22 person-rem, as indicated in Section 4.4.4.1.

As shown in Tables 4–24 and 4–25:

- 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 fatal cancer (or one in 5.3 billion years).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0031 person-rem per year, with an associated 1.6×10^{-6} latent cancer fatalities per year (or one in 625,000 years).

- 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 in 113 years).

- The dose to the maximally exposed offsite individual from six-month PUREX processing would be 0.00051 millirem, with an associated 2.6×10^{-10} risk of developing fatal cancer (or one chance in 3.8 billion).

Table 4–24 Annual Radiological Impacts to the Public From Operational Activities Associated With Alternative 3

Receptor	*Alternative 3*	
	PUREX Process of Declad and Cleaned Blanket Spent Nuclear Fuel at SRS [a]	***Declad and Clean Blanket Fuel and Treat Driver Spent Nuclear Fuel at ANL-W***
Population Dose Within 80 Kilometers (50 Miles) in the Year 2010		
Dose (person-rem)	0.02 [b]	0.0031
Latent cancer fatalities	0.000010	1.6×10^{-6}
Annual Dose to the Maximally Exposed Offsite Individual		
Dose (millirem)	0.00051 [b]	0.00038
Latent cancer fatality risk	2.6×10^{-10}	1.9×10^{-10}
Percent of natural background [c]	0.00017	0.00011
Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)		
Dose (millirem)	0.000024 [d]	0.000013 [e]
Latent cancer fatality risk	1.2×10^{-11}	6.6×10^{-12}

[a] Includes airborne and liquid dose components over the six-month processing duration.
[b] Liquid dose contributions to the population and the maximally exposed individual dose are 0.00068 person-rem and 0.00012 millirem, respectively.
[c] The annual natural background radiation level at INEEL and at SRS is 360 and 300 millirem, respectively, for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem at INEEL (254,000 at SRS) .
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of the SRS F-Canyon in the year 2010 (848,000).
[e] 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 4–25 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 3

Impact	*Alternative 3*	
	Operations at SRS [b]	*Operations at ANL-W*
Worker [a]		
Total dose (person-rem per year)	38 [c]	22
Fatal cancers	0.015 [c]	0.079 [d]
Average worker dose (millirem per year)	250 [c]	60
Fatal cancer risk	0.00010 [c]	0.00022 [d]

[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 N441.1.
[b] Estimates from DOE 1998g.
[c] Operations at SRS to treat blanket spent nuclear fuel at F-Canyon are performed over six months.
[d] Operations at ANL-W to declad and clean blanket spent nuclear fuels and treat driver spent nuclear fuels are performed over nine years.

- The collective dose to the population within 80 kilometers (50 miles) of the F-Canyon would be 0.02 person-rem, with an associated 0.000010 latent cancer fatalities (or one chance in 100,000).

- The collective dose to F-Canyon facility workers would be 38 person-rem, with an associated 0.015 latent cancer fatalities (or one chance in 67).

Hazardous Chemical Impacts

It is expected that hazardous chemical impacts associated with this alternative at ANL-W would be small, as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described in Section 3.2.10.2.

For SRS, both carcinogenic and noncarcinogenic health effects to the public were assessed from exposure to hazardous chemicals. It was assumed that under normal operating conditions, the primary exposure pathway for members of the public would be via air emissions.

The 24-hour concentrations provided in Section 4.5.1 were converted to annual concentrations by using the appropriate regulatory scaling factor of 0.125 based on South Carolina's Air Quality Modeling Guidelines (SCDHEC 1993). This annual concentration for each noncarcinogenic chemical was divided by the corresponding inhalation reference concentration to estimate the hazard quotient for each chemical. The hazard quotients were summed to give the hazard index from all noncarcinogenic chemicals for this alternative. A hazard index less than 1 indicates that adverse health effects from noncancer-causing agents are not expected. For carcinogens, the annual concentration was multiplied by the unit cancer risk to estimate the increased cancer risk from that chemical. Hazardous chemical health effects are summarized in **Tables 4–26** and **4–27**.

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 is performed in the Hot Fuel Examination Facility; treatment of driver spent nuclear fuel is performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing takes 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 is 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 seismic event was analyzed for the blanket spent nuclear fuel. Table 4–28 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 4–26 Hazardous Chemical Impacts to the Public From Operational Activities at SRS for Alternative 3

Chemical	*Annual Concentration (micrograms per cubic meter)*	*Reference Concentration Inhalation (micrograms per cubic meter)*	*Unit Cancer Risk (risk per microgram per cubic meter)*	*Hazard Quotient*	*Cancer Risk*
Benzene	1.4×10^{-6}	None	0.0078	None	1.1×10^{-8}
Ethyl benzene	1.3×10^{-6}	1	None	1.3×10^{-6}	None
Formaldehyde	1.3×10^{-6}	None	0.013	None	1.6×10^{-8}
Hexane	1.4×10^{-6}	0.2	None	7.1×10^{-6}	None
Manganese	1.3×10^{-6}	0.000050	None	0.025	None
Methyl ethyl ketone	2.5×10^{-6}	1	None	2.5×10^{-6}	None
Methylene chloride	7.1×10^{-7}	None	0.00047	None	3.3×10^{-10}
Naphthalene	1.3×10^{-6}	0.003	None	0.00042	None
Toluene	1.4×10^{-6}	0.4	None	3.5×10^{-6}	None
Vinyl acetate	1.3×10^{-6}	0.2	None	6.3×10^{-6}	None
			Hazard Index	0.025	None

Source: EPA 1999.

Table 4–27 Hazardous Chemical Impacts to the Noninvolved Worker From Operational Activities at SRS for Alternative 3

Chemical	*Annual Concentration (micrograms per cubic meter)*	*Reference Concentration-inhalation (micrograms per cubic meter)*	*Unit Cancer Risk (risk per microgram per cubic meter)*	*Hazard Quotient*	*Cancer Risk*
Benzene	0.0071	None	0.0078	None	0.000055
Ethyl benzene	0.0035	1	None	0.0035	None
Formaldehyde	0.0035	None	0.013	None	0.000046
Hexane	0.0071	0.2	None	0.035	None
Manganese	3.7×10^{-11}	0.00005	None	7.4×10^{-7}	None
Methyl ethyl ketone	0.0035	1	None	0.0035	None
Methylene chloride	0.0071	None	0.00047	None	3.3×10^{-6}
Naphthalene	1.5×10^{-11}	0.003	None	5.0×10^{-10}	None
Toluene	0.0071	0.4	None	0.018	None
Vinyl acetate	0.0035	0.2	None	0.018	None
			Hazard Index	0.078	NA

Sources: DOE 1998g, EPA 1999.

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 4–29.

Table 4–28 Accident Frequency and Consequences at ANL-W for 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)	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}
Earthquake (design-basis earthquake at Hot Fuel Examination Facility)	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Earthquake (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	2.4×10^{-3}	1.2×10^{-9}	2.8×10^{-4}	1.4×10^{-7}	4.9×10^{-5}	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}	0.689	3.4×10^{-4}	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 4–29 Annual Cancer Risks Due to Accidents at ANL-W for 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}	1.8×10^{-8}	3.4×10^{-12}
Earthquake (design-basis earthquake)	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Earthquake (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}	3.5×10^{-9}	8.7×10^{-11}
Sodium fire	2.3×10^{-8}	2.7×10^{-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 in 4.5 million years) and 1.5×10^{-8} per year (or one in 66.7 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one in 76,920 years).

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 4–30** 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 4–31**.

Table 4–30 Accident Frequency and Consequences at SRS for 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)*	*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}
Earthquake (F-Canyon)	0.00013	1100	0.00055	2100	1.1	12000	0.0048
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
Aircraft crash	NA	NA	NA	NA	NA	NA	NA

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
NA = Not analyzed, because the frequency is estimated to be less than 10^{-7} per year (see Appendix F for details).

Table 4–31 Annual Cancer Risks Due to Accidents at SRS for 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}
Earthquake (F-Canyon)	7.2×10^{-8}	0.00014	4.8×10^{-7}
Earthquake (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 in 13.9 million years) and 4.8×10^{-7} per year (or one in 2.1 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00017 per year (or one in 5,880 years).

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 4–32.

Table 4–32 Hazardous Chemical Impacts Due to Accidents at ANL-W for Alternative 3

Accident	*Frequency (event per year)*	*Receptor Location*	*Exposure*
Uranium-handling accident	0.01	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.008	Noninvolved worker at 100 meters	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker at 100 meters	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 Draft EIS (DOE 1998g) 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 4–33**, and are considered representative of wet storage accidents at SRS.

Table 4–33 Hazardous Chemical Impacts Due to Accidents at SRS for Alternative 3

Accident	*Frequency (event per year)*	*Receptor Location*	*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 at 640 meters	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.
ERPG = Emergency Response Planning Guideline.
Source: DOE 1998g.

4.5.5 Environmental Justice

As discussed in Section 4.5.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the six months of PUREX processing of blanket spent nuclear fuel at SRS and nine years of electrometallurgical treatment of driver spent nuclear fuel at ANL-W would be 1.7×10^{-9} (or one chance in 588 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000014 (or one chance in 71,400). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small regardless of the racial and ethnic composition of the population and independent of the economic status of individuals comprising the population. Operation of PUREX at SRS, and electrometallurgical treatment and decladding and cleaning processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

4.5.6 Waste Management

ANL-W

This alternative would generate process wastes from treatment operations, other associated process wastes from normal support operations, and deactivation wastes following the conclusion of operations. Process wastes would include fuel assembly hardware and metal and ceramic high-level radioactive wastes. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes 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 4–7 (see Section 4.2.6).

Estimates of the total amount of other associated process wastes generated as a result of Alternative 3 are provided in **Table 4–34**. These values 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 4–34 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.

Direct Process Wastes

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, metal 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 metal 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 metal waste form. Both the ceramic and metal waste would be categorized as

high-level radioactive waste. The volumes of waste forms provided in Table 4–34 are for the standardized canisters required for disposal of these materials.

Table 4–34 Amounts of Wastes Generated at ANL-W for Alternative 3[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Wastes		
Fuel assembly hardware (low-level radioactive waste)	37.5	13,100
High-level radioactive ceramic waste	16.3	24,400
High-level radioactive metal waste	2.0	2,500
Spent nuclear fuel	0	0
Other Associated Process Wastes		
High-level radioactive wastes	0.4	220
Low-level radioactive wastes[b]	555	113,000
Transuranic wastes	9.1	3,800
Mixed wastes	27.5	14,800
Sanitary wastes	4,960	1.72×10^{6}
Deactivation Wastes		
Low-level radioactive wastes[b]	178	60,000
Transuranic wastes	1.6	853
Mixed wastes	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 are accomplished during this time period.

[b] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

The metal and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel at ANL-W would be temporarily stored for 10 to 15 years 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, and shielding would be provided by a combination of steel storage liners storing the waste and by the soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste cans containing the metal 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

These high-level radioactive wastes 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 absorbant 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 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 Wastes

These low-level radioactive wastes 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 will 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 Wastes

These transuranic wastes would be generated by Alternative 3 from decontamination activities, repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic wastes 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 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 approximately 9 cubic meters (318 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 Wastes

These mixed wastes 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 1996b).

Deactivation Wastes

A variety of wastes 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. These would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes 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 wastes 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 wastes represent an additional 30 percent over the total incidental waste (excluding sanitary wastes) requiring disposal.

SRS

The PUREX process at SRS would generate process wastes from treatment operations and other associated process wastes from support operations. Process wastes would include high-level radioactive waste. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. The incidental wastes include low-level radioactive wastes, transuranic wastes, and mixed wastes. 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 4–35**.

Table 4–35 Summary of Waste Material Categories at SRS for Alternative 3

Waste Stream	*Category*	*Interim Storage*	*Final Disposal*
Process Wastes			
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.	Offsite (proposed geologic) repository
Other Associated Process Wastes			
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	Off site

Estimates of the amounts of wastes generated as a result of the PUREX alternative at SRS are provided in **Table 4–36**. 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 under the PUREX alternative.

As indicated in the following waste type discussions, the amounts of wastes associated with this processing alternative are relatively small compared to onsite and offsite management capacities.

Table 4–36 Amounts of Wastes Generated at SRS for Alternative 3

Waste Stream	*Total Waste Generated (cubic meters)*[a]
Direct Process Wastes	
Liquid high-level radioactive waste	510
Equivalent Defense Waste Processing Facility Canisters[b]	5.6 (9 canisters)
Saltstone[b]	1,290
Other Associated Process Wastes	
Low-level radioactive waste	3,600[c]
Transuranic waste	90
Mixed waste	6.9

[a] These values are estimated based on heavy metal mass ratio of similar materials processed at SRS (20 metric tons of heavy metal) and provided in DOE 1998g.
[b] These wastes result from processing the liquid high-level radioactive waste.
[c] Assuming a volume reduction factor of 4, the estimated disposal volume would be about 900 cubic meters (31,780 cubic feet).

Direct Process Wastes

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 an offsite (proposed 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.

Other Associated Process Low-Level Radioactive Wastes

These 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 Wastes

The volume of transuranic waste 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 Wastes

These mixed wastes 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 TREAT (ELECTROMETALLURGICAL TREATMENT) 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 fuels 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 be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel could start in 2003 and subsequent treatment by melt and dilute at ANL-W could start in 2005 and be completed by 2012.

4.6.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that this alternative at ANL will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

Radiological Gaseous Emissions

Cleaning blanket spent nuclear fuel and electrometallurgical treatment of 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 occurs when electrometallurgical treatment processing of driver spent nuclear fuels performed simultaneously with cutting blanket spent nuclear fuels for sodium removal prior to the melt and dilute process. This simultaneous operation would occur over a 3-year period during the estimated 10 years of operation, starting in 2003. 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. 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 Big Lost River are not 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 NPDES stormwater and liquid effluent discharges associated with facilities at ANL-W.

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 1996a).

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 4–37** and **4–38** for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in **Tables 4–39** and **4–40**. The impacts from hazardous chemical releases during accident conditions are presented in **Table 4–41**. 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 fuels are 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 radiological impacts to the public are given in Table 4–37. The impacts are calculated for two receptor groups: 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 gases (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.

Table 4–37 Annual Radiological Impacts to the Public From Operational Activities Associated With Alternative 4

Receptor	*Alternative 4*		
	Electrometallurgical Treatment of Driver Spent Nuclear Fuel	*Clean and Melt and Dilute Blanket Spent Nuclear Fuel at ANL-W*	*Total*
Population Dose Within 80 Kilometers (50 Miles) in the Year 2010			
Dose (person-rem)	0.0028	0.00028	0.0031
Latent cancer fatalities	1.4×10^{-6}	1.4×10^{-7}	1.6×10^{-6}
Annual Dose to the Maximally Exposed Offsite Individual			
Dose (millirem)	0.00033	0.000048	0.00038
Latent cancer fatality risk	1.6×10^{-10}	2.4×10^{-11}	1.9×10^{-10}
Percent of natural background[a]	0.000092	0.000013	0.00011
Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)[b]			
Dose (millirem)	0.000012	1.2×10^{-6}	0.000013
Latent cancer fatality risk	6.0×10^{-12}	6.0×10^{-13}	6.6×10^{-12}

[a] The annual natural background radiation level at INEEL is 360 millirem for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.

[b] 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 4–38 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 total worker population 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, the cumulative worker population dose would be 286 person-rem, leading to a risk of 0.11 latent cancer fatalities.

Table 4–38 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 4

Impact	*Alternative 4*
Worker[a]	
Total dose (person-rem per year)	22
13-year fatal cancer risk	0.11
Average worker dose (millirem per year)	60
13-year fatal cancer risk	0.00031

[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 N441.1.

Source: ANL 1999.

As shown in Tables 4–37 and 4–38:

- 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 fatal cancer (or one in 5.3 billion years).
- The collective dose to the population within the 80 kilometers (50 miles) of the ANL-W facilities would be 0.0031 person-rem per year, with an associated 1.6×10^{-6} latent cancer fatalities per year (or one in 625,000 years).
- 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 in 113 years).

Hazardous Chemical Impacts

It is expected that hazardous chemical impacts associated with this alternative at ANL-W will be small as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described 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 is performed in the Hot Fuel Examination Facility; treatment of driver spent nuclear fuel is performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing takes 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 seismic event. The melt and dilute processing of blanket spent nuclear fuel is performed only in the Hot Fuel Examination Facility. Melt and dilute processing of the fuel results 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 the one facility. Therefore, only the higher frequency design-basis seismic event was analyzed. Table 4–39 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 4–40.

Table 4–39 Accident Frequency and Consequences for 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)*	*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}
Earthquake (design-basis earthquake at Hot Fuel Examination Facility)	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9
Earthquake (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}	0.689	0.00034	0.054	2.2×10^{-8}
Earthquake (design-basis event)	0.008	471	0.00024	56.1	0.028	15.2	6.1×10^{-6}
Waste-handling spill	0.024	15	7.5×10^{-6}	1.77	0.00089	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 4–40 Annual Cancer Risks Due to Accidents at ANL-W for 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}	1.8×10^{-8}	3.4×10^{-12}
Earthquake (design-basis earthquake)	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Earthquake (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}	3.5×10^{-9}	8.7×10^{-11}
Sodium Fire	2.3×10^{-8}	2.7×10^{-6}	1.7×10^{-10}
Earthquake (design-basis event)	1.9×10^{-6}	0.00022	4.9×10^{-8}
Waste-handling spill	1.8×10^{-7}	0.000021	4.7×10^{-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 1.9×10^{-6} per year (or one in 526,300 years) and 4.9×10^{-8} per year (or one in 20.4 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00022 per year (or one in 4,545 years).

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 4–41.

4.6.5 Environmental Justice

As discussed in Section 4.6.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the 13 years of melt and dilute processing of blanket spent nuclear fuel and electrometallurgical or melt and dilute treatment of driver spent nuclear fuels would be 2.5×10^{-9} (or one chance in 400 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000021 (or one chance in 47,600). Radiological and nonradiological risks posed by implementation of this alternative would therefore be small regardless of the racial and ethnic composition of the population, and independent of the economic status of individuals comprising the population. Operation of electrometallurgical and melt and dilute treatment processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

Table 4–41 Nonradiological Impacts of Accidents for Alternative 4

Accident	*Frequency event/year*	*Receptor Location*	*Exposure*
Uranium-handling accident	0.01	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.00001	Noninvolved worker at 100 meters	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker at 100 meters	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

4.6.6 Waste Management

This alternative would generate process wastes from treatment operations, other associated process wastes from normal support operations, and wastes following the conclusion of operations. Process wastes would include metal and ceramic high-level radioactive wastes. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes 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 4–7 (see Section 4.2.6).

Estimates of the total amount of other associated process wastes generated as a result of Alternative 4 are provided in **Table 4–42**. These values 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 4–42 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 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.

Direct Process Wastes

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, metal 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 metal 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 metal waste form. Both the ceramic and metal waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–42 are for the standardized canisters required for disposal of these materials.

Table 4–42 Amounts of Wastes Produced at ANL-W for Alternative 4[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Wastes		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	16.3	24,400
High-level radioactive metal waste	2.0	2,500
Melt and dilute product	45.6	114,000
Other Associated Process Wastes		
High-level radioactive wastes	0.4	220
Low-level radioactive wastes[b]	650	132,000
Transuranic wastes	11.2	4,730
Mixed wastes	32.1	17,300
Sanitary wastes	4,960	1.72×10^{6}
Deactivation Wastes		
Low-level radioactive wastes[b]	178	66,000
Transuranic wastes	1.6	853
Mixed wastes	5.1	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 are accomplished during this time period.

[b] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

The metal and ceramic high-level radioactive waste and the melted blanket spent nuclear fuel generated at ANL-W would be temporarily stored for 10 to 15 years 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, and shielding would be provided by a combination of steel storage liners storing the waste and the shielding provided by the soil surrounding the liners. When an offsite (proposed 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

These high-level radioactive wastes 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 absorbant 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 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 Wastes

These low-level radioactive wastes 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 driver spent nuclear fuel at ANL-W that will 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 Wastes

These transuranic wastes would be generated by Alternative 4 from decontamination activities, repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic wastes 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 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 is approximately 9 cubic meters (318 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 Wastes

These mixed wastes 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 1996b).

Deactivation Wastes

A variety of wastes would be generated as part of deactivation activities associated with processing at ANL-W. These would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes 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 wastes 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. If the deactivation waste volume is generated in a single year, the wastes would represent an increase of approximately three times the annual waste generated by the treatment operations of Alternative 4. The total deactivation wastes represent an additional 30 percent over the total incidental waste requiring disposal.

4.7 ALTERNATIVE 5: DECLAD AND CLEAN BLANKET FUEL AND TREAT (ELECTROMETALLURGICAL TREATMENT) 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 fuels 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 be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its shipment to SRS could start in 2003 and be completed by 2009. Current planning at SRS has scheduled the melt and dilute process at Building 105-L for other missions (DOE 1998g). Melt and dilute process of blanket spent nuclear fuel at SRS could start around 2020, if capacity becomes available, and be completed by 2023.

4.7.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that this alternative at ANL-W will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

At SRS, nonradiological air emissions result from operation of ancillary support facilities for the melt and dilute process at Building 105-L. These include: site electrical power generators, emergency diesel generators, fuel handling activities in the L-area basin, and increased vehicle emissions. The largest contributors to the emissions are the onsite electrical power generators (Bickford 1999).

The concentrations of nonradiological air pollutants attributed to this alternative at SRS are presented in **Table 4–43**, along with the total estimated site air pollutant concentrations. The concentrations for the alternative are based on information in the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g). The concentrations have been adjusted to account for the increased mass of sodium-bonded spent nuclear fuel. The total concentrations are equal to the concentration for the alternative plus the baseline concentrations from Section 3.3.3.1. The concentrations are compared to their corresponding ambient air quality standards. 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. Health effects from hazardous chemicals associated with this alternative are addressed in Section 4.7.4.1.

Table 4–43 Nonradiological Air Quality Concentrations Associated with Alternative 5 at SRS for Comparison with Ambient Air Quality Standards

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*	*Alternative 5 Concentration (micrograms per cubic meter)*	*Total Site Concentration (micrograms per cubic meter)*
Criteria Pollutants				
Carbon monoxide	8 hours 1 hour	10,000 40,000	0.08 0.51	631.88 5015.11
Nitrogen dioxide	Annual	100	less than 0.01	8.8
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 150 150	ND ND ND	4.8 80.6 NA
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 65	NA NA	NA NA
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.01 0.03 ND	16.31 215.43 690.2
State-Regulated Pollutants				
Gaseous fluoride	30 days 7 days 24 hours 12 hours	0.8 1.6 2.9 3.7	ND ND ND ND	0.11 0.11 0.60
Total suspended particulates	Annual	75	less than 0.01	43.3
Hazardous and Other Toxic Compounds				
1,1,1-trichloroethane	24 hours	9,550	less than 0.01	less than 22.01
Benzene	24 hours	150	ND	31
Ethanolamine	24 hours	200	less than 0.01	less than 0.02
Ethyl benzene	24 hours	4,350	ND	0.12
Ethylene glycol	24 hours	650	less than 0.01	less than 0.09
Formaldehyde	24 hours	15	less than 0.01	less than 0.02
Glycol ethers	24 hours	No Standard	less than 0.01	less than 0.02
Hexachloronaphthalene	24 hours	1	less than 0.01	less than 0.02
Hexane	24 hours	900	less than 0.01	less than 0.08
Manganese	24 hours	25	ND	0.1
Methyl alcohol	24 hours	1,310	less than 0.01	less than 0.52
Methyl ethyl ketone	24 hours	14,750	less than 0.01	less than 1
Methyl isobutyl ketone	24 hours	2,050	ND	less than 0.51
Methylene chloride	24 hours	8,750	ND	1.8
Naphthalene	24 hours	1,250	less than 0.01	less than 0.02
Nitric acid	24 hours	125	ND	6.7
Phenol	24 hours	190	ND	0.03
Phosphorous	24 hours	0.5	ND	less than 0.001
Sodium hydroxide	24 hours	50	ND	0.01
Toluene	24 hours	2,000	less than 0.01	less than 1.61
Trichloroethane	24 hours	6,750	ND	1
Vinyl acetate	24 hours	176	ND	less than 0.02
Xylene	24 hours	4,350	less than 0.01	less than 3.81

NA = Not Available; ND=not detectable.
Source: Bickford et al. 1997, plus baseline concentrations from Section 3.3.3.1.

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 are the most prevalent radioactive gaseous fission products that would be 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 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 occurs when decladding the blanket spent nuclear fuel for packaging and shipment to SRS and electrometallurgical treatment of driver spent nuclear fuels are performed simultaneously. This simultaneous operation was estimated to occur over a three-year period starting in 2003. Based on an annual decladding throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and an electrometallurgical treatment process 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.

Since declad and clean fuels 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 fuels 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 fuels and electrometallurgical treatment of driver spent nuclear fuels 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 treating blanket spent nuclear fuels 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 (G-10). Since employment would not increase as a result of processing these fuels, 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 1998g).

Radiological Liquid Effluent

There are no anticipated radiological liquid effluents associated with 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 4–44 through 4–46 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–47 through 4–50. 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 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 at ANL-W would result when decladding and cleaning of blanket spent nuclear fuels and treatment of driver spent nuclear fuels are performed simultaneously under this alternative. The doses from decladding and cleaning blanket spent nuclear fuels and treating driver spent nuclear fuels at ANL-W would be similar to those presented for Alternative 2 in Section 4.4.4.1.

Calculated maximum annual radiological impacts to the public are given in **Table 4–44**. The impacts are calculated for two receptor groups: 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 are releases of tritium gases (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.

Table 4–44 Annual Radiological Impacts to the Public from Operational Activities Associated With Alternative 5

	Alternative 5	
Receptor	*Melt and Dilute Blanket Spent Nuclear Fuel at SRS*	*Clean Blanket Spent Nuclear Fuel and Electrometallurgical Treatment of Driver Spent Nuclear Fuel at ANL-W*
Population Dose Within 80 Kilometers (50 Miles) in the Year 2010		
Dose (person-rem)	0.0076	0.0031
Latent cancer fatalities	3.8×10^{-6}	1.5×10^{-6}
Annual Dose to the Maximally Exposed Offsite Individual		
Dose (millirem)	0.00010	0.00038
Latent cancer fatality risk	5.0×10^{-11}	1.9×10^{-10}
Percent of natural background[a]	0.000033	0.000011
Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)		
Dose (millirem)	0.000011 [b]	0.000013 [c]
Latent cancer fatality risk	5.5×10^{-12}	6.6×10^{-12}

[a] The annual natural background radiation level at INEEL and at SRS is 360 and 300 millirem, respectively, for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 and 215,000 person-rem, respectively.

[b] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of SRS Building 105-L (L-Reactor Area) in the year 2010 (715,000).

[c] 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).

The blanket spent nuclear fuel has been declad and cleaned at ANL-W, where it is expected that the gaseous fission products would have been released. However, for the melt and dilute process, it is conservatively assumed that these gaseous fission products are 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 4–45 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 total worker population 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 cumulative worker population dose would be 150 person-rem, leading to a risk of 0.06 latent cancer fatalities. The estimated annual total worker population dose to declad and clean blanket spent nuclear fuels and treat driver spent nuclear fuels at ANL-W is 22 person rem, as indicated in Section 4.4.4.1.

Table 4–45 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 5

Impact	*Alternative 5*	
	Operations at SRS	*Operations at ANL-W*
Worker[a]		
Total dose (person-rem per year)	50	22
Fatal cancer risk	0.06 [b]	0.079 [c]
Average worker dose (millirem per year)	500	60
Fatal cancer risk	0.00060[b]	0.00022[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 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 N441.1.

[b] Operations at SRS to treat blanket spent nuclear fuel using melt and dilute at Building 105-L are performed over three years.

[c] Operations at ANL-W to declad and clean blanket spent nuclear fuels and treat driver spent nuclear fuels are performed over nine years.

Sources: ANL 1999, DOE 1998g.

As shown in Tables 4–44 and 4–45:

- 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 fatal cancer (or one in 5.3 billion years).

- The collective dose to the population within the 80 kilometers (50 miles) of the ANL-W facilities would be 0.0031 person-rem per year, with an associated 1.6×10^{-6} latent cancer fatalities per year (or one in 625,000 years).

- 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 in 113 years).

- 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 5.0×10^{-11} risk per year of developing fatal cancer (or one in 20 billion years).

- The annual 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 in 263,100 years).

- 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 in 50 years).

Hazardous Chemical Impacts

It is expected that the hazardous chemical impacts associated with this alternative at ANL-W will be small, as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described in Section 3.2.10.2.

For SRS, both carcinogenic and noncarcinogenic health effects to the public were assessed from exposure to hazardous chemicals. It was assumed that, under normal operating conditions, the primary exposure pathway for members of the public would be via air emissions.

The 24-hour concentrations provided in Section 4.7.1 were converted to annual concentrations by using the appropriate regulatory scaling factor of 0.125 based on South Carolina's Air Quality Modeling Guidelines (SCDHEC 1993). The annual concentration for each noncarcinogenic chemical was divided by the corresponding inhalation reference concentration to estimate the hazard quotient for each chemical. The hazard quotients were summed to give the hazard index from all noncarcinogenic chemicals for this alternative. A hazard index less than one indicates that adverse health effects from noncancer-causing agents are not expected. For carcinogens, the annual concentration was multiplied by the unit cancer risk to estimate the increased cancer risk from that chemical. Hazardous chemical health effects to the public and noninvolved worker are summarized in **Table 4–46** and **4–47**, respectively.

Table 4–46 Hazardous Chemical Impacts to the Public from Operational Activities at SRS for Alternative 5

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	ND	None	0.0078	None	ND
Ethyl benzene	ND	1	None	ND	None
Formaldehyde	1.3×10^{-6}	None	0.013	None	1.6×10^{-8}
Hexane	1.3×10^{-6}	0.2	None	6.3×10^{-6}	None
Manganese	ND	0.00005	None	ND	None
Methyl ethyl ketone	1.3×10^{-6}	1	None	1.3×10^{-6}	None
Methylene chloride	ND	None	0.00047	None	ND
Naphthalene	1.3×10^{-6}	0.003	None	0.00042	None
Toluene	1.3×10^{-6}	0.4	None	3.1×10^{-6}	None
Vinyl acetate	ND	0.2	None	ND	None
			Hazard Index	0.00043	None

Source: EPA 1999.
Key: ND = Not detectable.

Table 4–47 Hazardous Chemical Impacts to the Noninvolved Worker from Operational Activities at SRS for Alternative 5

Chemical	*Annual Concentration (milligrams per cubic meter)*	*Reference Concentration Inhalation (milligram per cubic meter)*	*Unit Cancer Risk (risk per milligram per cubic meter)*	*Hazard Quotient*	*Cancer Risk*
Benzene	ND	None	0.0078	None	ND
Ethyl benzene	ND	1	None	ND	None
Formaldehyde	0.0035	None	0.013	None	0.000046
Hexane	0.0035	0.2	None	0.018	None
Manganese	ND	0.000050	None	ND	None
Methyl ethyl ketone	0.0035	1	None	0.0035	None
Methylene chloride	ND	None	0.00047	None	ND
Naphthalene	1.9×10^{-12}	0.003	None	6.3×10^{-10}	None
Toluene	0.0035	0.4	None	0.0088	None
Vinyl acetate	ND	0.2	None	ND	None
			Hazard Index	0.00043	None

Source: EPA 1999.
Key: ND = Not detectable above background levels.

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 is 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 takes 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 declading and cleaning of the blanket spent nuclear fuel is 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 seismic event was analyzed. **Table 4–48** 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 95^{th} percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50^{th} 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 4–49**.

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 in 4.5 million years) and 1.5×10^{-8} per year (or one in 66.7 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one in 76,920 years).

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 4–50** 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 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 4–48 Accident Frequency and Consequences at ANL-W for 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)	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}
Earthquake (design-basis earthquake at Hot Fuel Examination Facility)	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Earthquake (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}	0.689	0.00034	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 4–49 Annual Cancer Risks Due to Accidents at ANL-W for 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}	1.8×10^{-8}	3.4×10^{-12}
Earthquake (design-basis earthquake)	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Earthquake (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}	3.5×10^{-9}	8.7×10^{-11}
Sodium fire	2.3×10^{-8}	2.7×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

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 4–51**.

Table 4–50 Accident Frequency and Consequences at SRS for 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)	Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Waste-handling spill	0.024	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
Loss of cooling water	0.05	120	0.000060	500	0.25	1.3	5.2×10^{-7}
Fire	0.075	86	0.000043	140	0.07	6.3	2.5×10^{-6}
Criticality	NA	NA	NA	NA	NA	NA	NA
Aircraft crash	NA	NA	NA	NA	NA	NA	NA

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
Key: NA = Not analyzed, because the frequency is estimated to be less than 10^{-7} per year (see Appendix F for details).

Table 4–51 Annual Cancer Risks of Accidents at SRS for Alternative 5

Accident	Maximally Exposed Offsite Individual [a]	Population within 80 kilometers (50 miles) [b]	Noninvolved Worker [a]
Waste-handling spill	2.6×10^{-8}	0.000043	1.6×10^{-9}
Loss of power	6.6×10^{-6}	0.011	3.4×10^{-7}
Loss of cooling water	3.0×10^{-6}	0.013	2.6×10^{-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 in 151,500 years) and 3.4×10^{-7} per year (or one in 2.9 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.013 per year (or one in 77 years).

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: Electrometallurgical treatment of driver spent nuclear fuel at ANL-W and PUREX processing of blanket spent nuclear fuel at SRS.

4.7.5 Environmental Justice

As discussed in Section 4.7.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the three years of melt and dilute treatment of blanket spent nuclear fuel at SRS and the nine years of combined operations to clean blanket spent nuclear fuels and electrometallurgically treat driver spent nuclear fuels at ANL-W would be 1.7×10^{-9} (or one chance in 588 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000014 (or one chance in 74,100). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small, regardless of the racial and ethnic composition of the population and independent of the economic status of individuals comprising the

population. Operation of melt and dilute treatment at SRS and electrometallurgical processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

4.7.6 Waste Management

ANL-W

This alternative would generate process wastes from treatment operations, other associated process wastes from normal support operations, and deactivation wastes following the conclusion of operations. Process wastes would include fuel hardware and metal and ceramic high-level radioactive wastes. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes 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 4–7 (see Section 4.2.6).

Estimates of the total amount of other associated process wastes generated as a result of Alternative 5 are provided in **Table 4–52**. These values 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 4-52 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 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.

Table 4–52 Amounts of Wastes Produced at ANL-W for Alternative 5[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Wastes		
Fuel assembly hardware (low-level radioactive waste)	37.5	13,100
High-level radioactive ceramic waste	16.3	24,400
High-level radioactive metal waste	2.0	2,500
Spent nuclear fuel	0	0
Other Associated Process Wastes		
High-level radioactive wastes	0.4	220
Low-level radioactive wastes[b]	555	113,000
Transuranic wastes	5.1	3,800
Mixed wastes	27.5	14,800
Sanitary wastes	4,960	1.7×10^6
Deactivation Wastes		
Low-level radioactive wastes[b]	178	60,000
Transuranic wastes	1.6	853
Mixed wastes	9.1	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 are accomplished during this time period.

[b] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Direct Process Wastes

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, metal 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 metal 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 metal waste form. Both the ceramic and metal waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–52 are for the standardized canisters required for disposal of these materials.

The metal and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel at ANL-W would be temporarily stored for 10 to 15 years 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, and shielding would be provided by a combination of steel storage liners storing the waste and the shielding provided by the soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste cans containing the metal 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 Wastes

These high-level radioactive wastes 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 absorbant 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 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 Wastes

These low-level radioactive wastes 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 will 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 Wastes

These transuranic wastes would be generated by Alternative 5 from decontamination activities, repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic wastes 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 is approximately 9 cubic meters (318 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 Wastes

These mixed wastes 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 1996b).

Deactivation Wastes

A variety of wastes would be generated as part of deactivation activities at ANL-W. These would include process equipment and process material such as electrorefiner salt and cadmium from electrometallurgical treatment of driver spent nuclear fuel. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes 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 wastes would be low-level radioactive wastes 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 is generated over a period of two years. The total deactivation wastes represent an additional 30 percent over the total incidental waste requiring disposal.

SRS

The melt and dilute process at SRS would generate process wastes from treatment operations and other associated process wastes from support operations. Process wastes would include metallic high-level radioactive waste. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. The incidental wastes include low-level radioactive wastes, transuranic wastes, and mixed wastes. 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 4–53**.

Table 4–53 Summary of Waste Material Categories at SRS for Alternative 5

Waste Stream	*Category*	*Interim Storage*	*Final Disposal*
Process Wastes			
Metallic waste form Off-gas filters	Conditioned spent nuclear fuel High-level radioactive waste	L-Area L-Area	Offsite (proposed geologic) repository Offsite (proposed geologic) repository
Other Associated Process Wastes			
Less than 100 nanocuries per gram transuranic waste	Low-level radioactive waste	None	Low-activity waste vaults Waste Isolation Pilot Plant
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Transuranic waste storage pads	Offsite
Contaminated	Mixed waste	Mixed waste storage buildings	Offsite

Estimates of the amounts of wastes generated as a result of the melt and dilute alternative at SRS are provided in **Table 4–54**. 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 under the melt and dilute alternative.

Table 4–54 Amounts of Wastes Generated at SRS for Alternative 5

Waste Stream	*Total Waste Generated (cubic meters)*[a]
Direct Process Wastes	
Canisters of melt and dilute products	76 (189 canisters)
Liquid high-level radioactive wastes[b]	30[c]
Saltstone[b]	78
Other Associated Process Wastes	
Low-level radioactive waste	1,320[d]
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 1998g.

[b] These are secondary process wastes, high-level radioactive wastes.

[c] This is a liquid high-level radioactive waste volume which results in about one Defense Waste Processing facility borosillicate glass high-level radioactive waste canister or a solid high-level radioactive waste volume of 0.62 cubic meters.

[d] Assuming a volume reduction factor of 4, the estimated disposal volume would be about 330 cubic meters (11,650 cubic feet).

As indicated in the following waste-type discussions, the amounts of wastes associated with this processing alternative are relatively small compared to onsite and offsite management capacities.

Direct Process Wastes

During the melt and dilute process, high-level radioactive waste ingots 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 wastes are 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 an offsite (proposed 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 Wastes

These low-level radioactive wastes 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 Wastes

The volume of transuranic waste 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 Wastes

These mixed wastes 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 products produced from 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 fuels 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 treatment of driver and blanket spent nuclear fuel by melt and dilute at ANL-W could start in 2005 and could be completed by 2015.

4.8.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that this alternative at ANL will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

Radiological Gaseous Emissions

The cleaning of the blanket and driver spent nuclear fuels and the melt and dilute treatment of these fuels 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 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 release of radioactive gases occurs when chopping of blanket and driver spent nuclear fuels to remove metallic sodium is performed simultaneously. This simultaneous operation could occur over a two-year period during the estimated 10 years of operation, starting in 2003. 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.

4.8.2 Water Resources

Surface Water

No surface water is used at ANL-W. The facilities at ANL-W are not expected to be reached by flood waters from Big Lost River, 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 NPDES stormwater and liquid effluent discharges associated with facilities at ANL-W.

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 to prevent hazardous materials from release to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996a).

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 4–55 and 4–56 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–57 and 4–58. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–59. 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 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 fuels 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 radiological impacts to the public are given in **Table 4–55**. The impacts are calculated for two receptor groups: 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 gases (about 1 percent of which were conservatively assumed 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 are also included in the table.

Table 4–55 Annual Radiological Impacts to the Public from Operational Activities Associated With Alternative 6

Receptor	*Alternative 6*		
	Melt and Dilute Blanket Spent Nuclear Fuel at ANL-W	*Melt and Dilute Driver Spent Nuclear Fuel at ANL-W*	*Total*
Population Dose Within 80 Kilometers (50 Miles) in the Year 2010			
Dose (person-rem)	0.00028	0.012	0.012
Latent cancer fatalities	1.4×10^{-7}	6.0×10^{-6}	6.0×10^{-6}
Annual Dose to the Maximally Exposed Offsite Individual			
Dose (millirem)	0.000048	0.0019	0.002
Latent cancer fatality risk	2.4×10^{-11}	9.5×10^{-10}	1.0×10^{-7}
Percent of natural background[a]	0.000013	0.00053	0.00054
Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)[b]			
Dose (millirem)	1.2×10^{-6}	0.00005	0.000051
Latent cancer fatality risk	6.0×10^{-13}	2.5×10^{-11}	2.6×10^{-11}

[a] The annual natural background radiation level at INEEL is 360 millirem for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.

[b] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of the ANL-W in the year 2010 (240,338).

Table 4–56 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 total worker population 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 out over the 10 years of treatment activities, the cumulative worker population dose would be 220 person-rem, leading to a risk of 0.088 latent cancer fatalities.

Table 4–56 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 6

Impact	*Alternative 6*
Worker[a]	
Total dose (person-rem per year)	22
10-year fatal cancer risk	0.088
Average worker dose (millirem per year)	60
10-year fatal cancer risk	0.00024

[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 N441.1.

Source: ANL 1999.

As shown in Tables 4-55 and 4-56:

- The annual dose to the maximally exposed offsite individual would be 0.00074 millirem per year, with an associated 3.7×10^{-10} risk per year of developing fatal cancer (or one in 2.7 billion years).

- The collective dose to the population within 80 kilometers of the ANL-W facilities would be 0.0044 person-rem per year, with an associated 2.2×10^{-6} latent cancer fatalities per year (or one in 454,000 years).

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities (or one in 113 years).

Hazardous Chemical Impacts

It is expected that hazardous chemical impacts associated with this alternative at ANL-W will be small, as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described 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 fuels is performed in the Hot Fuel Examination Facility. The multi-facility 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 seismic event was analyzed. **Table 4–57** 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 4–58**.

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 in 13,160 years) and 2.7×10^{-6} per year (or one in 370,400 years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0088 per year (or one in 113 years).

Table 4–57 Accident Frequency and Consequences for Alternative 6

Accident	Frequency (event/yr)	Maximally Exposed Offsite Individual		Population within 80 km (50 mi)		Noninvolved Worker	
		Dose (mrem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Latent Cancer Fatalities [b]	Dose (mrem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Waste-handling spill	0.024	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}
Earthquake (DBE)	0.008	19000	0.0095	2200	1.1	840	0.00034
Sodium Fire[c]	0.008	282	0.00014	33	0.016	2.6	1.0×10^{-6}
Criticality	0.003	0.52	2.6×10^{-7}	0.085	0.000043	0.47	1.9×10^{-7}
Blanket Spent Nuclear Fuel							
Waste-handling spill	0.024	14.9	7.5×10^{-6}	1.77	0.00089	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}
Earthquake (DBE)	0.008	472	0.00024	56.1	0.028	15.3	6.1×10^{-6}
Sodium Fire [c]	0.008	5.9	3.0×10^{-6}	0.689	0.00034	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.
Key: DBE = design-basis earthquake; km = kilometers; mi = miles; mrem = millirem; yr = year.

Table 4–58 Annual Cancer Risks Due to Accidents at ANL-W for Alternative 6

Accident	Maximally Exposed Offsite Individual [a]	Population within 80 km (50 mi) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Waste Liquid Spill	7.2×10^{-6}	0.00084	2.6×10^{-7}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask Drop	1.5×10^{-10}	1.8×10^{-8}	3.4×10^{-12}
Earthquake (DBE)	0.000076	0.0088	2.7×10^{-6}
Sodium Fire	1.1×10^{-6}	0.00013	8.0×10^{-9}
Criticality	7.8×10^{-10}	1.3×10^{-7}	5.7×10^{-10}
Blanket Spent Nuclear Fuel			
Waste Liquid Spill	1.8×10^{-7}	0.000021	4.8×10^{-9}
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.5×10^{-9}	8.7×10^{-11}
Earthquake (DBE)	1.9×10^{-6}	0.00022	4.9×10^{-8}
Sodium Fire	2.3×10^{-8}	2.7×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
Key: DBE = design-basis earthquake; km = kilometers; mi = miles.

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 fuels using the electrometallurgical process are summarized in **Table 4–59**.

Table 4–59 Hazardous Chemical Impacts Due to Accidents at ANL-W for Alternative 6

Accident	*Frequency (event per year)*	*Receptor Location*	*Exposure*
Sodium fire	0.008	Noninvolved Worker at 100 meters	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1

4.8.5 Environmental Justice

As discussed in Section 4.8.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the 10 years of melt and dilute treatment of sodium-bonded spent nuclear fuel at ANL-W would be 3.7×10^{-9} (or one chance in 270 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000022 (or one chance in 45,500). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small, regardless of the racial and ethnic composition of the population and independent of the economic status of individuals comprising the population. Operation of melt and dilute treatment processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

4.8.6 Waste Management

This alternative would generate process wastes from treatment operations, other associated process wastes from normal support operations, and deactivation wastes following the conclusion of operations. Process 0wastes would include metal and ceramic high-level radioactive wastes from stabilizing the residual wastes from the existing Electrometallurgical Demonstration Project. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes 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 4–7 (see Section 4.2.6).

Estimates of the total amount of other associated process waste generated as a result of Alternative 6 are provided in **Table 4–60**. These values 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 4–60 are for disposal and include volume reduction. It is anticipated that a large fraction of the low-level radioactive waste 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 4–60 Amounts of Wastes Produced at ANL-W for Alternative 6[a]

Waste Stream	Waste Quantities	
	Volume (m^3)	Mass (kg)
Direct Process Wastes		
Fuel assembly hardware (low-level radioactive wastes)	0	0
High-level radioactive ceramic waste	19.4	29,000
High-level radioactive metal waste	0.6	460
Melt and dilute product	65.6	136,400
Other Associated Process Wastes		
High-level radioactive wastes	0.4	220
Low-level radioactive wastes[b]	711	144,000
Transuranic wastes	12.5	5,400
Mixed wastes	35.3	19,000
Sanitary wastes	4,960	1.72×10^6
Deactivation Wastes		
Low-level radioactive wastes[b]	213	72,000
Transuranic wastes	1.6	853
Mixed wastes	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 are accomplished during this time period.

[b] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Direct Process Wastes

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, metal and ceramic high-level radioactive waste would be produced from existing process material at ANL-W. These wastes would be generated to stabilize materials produced during the demonstration. 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 4–60 are for the standardized canisters required for disposal of these materials.

A second metal high-level radioactive waste would be generated as a result of the melt and dilute treatment of fuel at ANL-W. It would be temporarily stored for 10 to 15 years 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, and shielding would be provided by a combination of steel storage liners and the shielding provided by the soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste cans containing the two metals and ceramic high-level radioactive wastes 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

These high-level radioactive wastes could 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 absorbant 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 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 Wastes

These low-level radioactive wastes 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 will 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 Wastes

These transuranic wastes would be generated by Alternative 6 from decontamination activities, repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic wastes 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 is approximately 13 cubic meters (459 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 Wastes

These mixed wastes 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 1996b).

Deactivation Wastes

A variety of wastes would be generated as part of deactivation activities associated with melt and dilute treatment of sodium-bonded spent nuclear fuel at ANL-W. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes 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 wastes would be low-level radioactive wastes generated as a result of dismantling and disposal 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 wastes represent approximately 30 percent over the total incidental 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 binned 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 bins. 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 4–61** lists the fuels that could be shipped as part of the applicable alternatives used to treat sodium-bonded spent nuclear fuel.

Table 4–61 Transportation Summary for Sodium-Bonded Fuels

Fuel Type	*Applicable 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/Onsite, intrafacility transfers
EBR-II driver	1,2,3,4,5,6	2.0	INTEC/ID	ANL-W/ ID	TN-FSV NAC-LWT	17/Onsite with roads open 43/Onsite with roads open
EBR-II blanket	All	22.4	ANL-W/ID	ANL-W/ ID	HFEF-5	165/Onsite, intrafacility transfers
Fast Flux Test Facility Driver	All	0.25	Hanford/WA	ANL-W/ID	T-3	10/ Public highways
Fermi-1 blanket	All	34.2	INTEC/ID	ANL-W/ID	PB-1	14/Onsite with road closed
Miscellaneous	All	0.04	Oak Ridge National Laboratory/TN Sandia National Laboratories/ NM	ANL-W/ID	To be determined by DOE	1/ Public highways 1/ Public highways
Declad EBR-II blanket	3,5	22.4	ANL-W	SRS/SC	NAC-LWT	11/Public highways
Declad Fermi-1 blanket	3,5	34.2	ANL-W	SRS/SC	NAC-LWT	18/Public highways

[a] "All" Includes the six alternatives plus the No Action Alternative.
Key: ID = Idaho; NM = New Mexico; SC = South Carolina; TN = Tennessee; WA = Washington.

The transportation of Fast Flux Test Facility driver spent nuclear fuel currently stored at the Hanford site and miscellaneous spent nuclear fuel currently stored at Oak Ridge National Laboratory and at Sandia National Laboratories are shipment campaigns related to sodium-bonded spent nuclear fuel and were analyzed by DOE in other NEPA documents, so they are not treated in detail 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 consequence of potential accidents during movement are bounded in frequency and consequence by handling accidents.

Fermi 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 consequence of any foreseeable accident are 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 consequence of any foreseeable accident are very small.

The following provides a summary of transportation impacts. Appendix G details the methods and assumptions used.

4.9.1 No Action Alternative Impacts

Under the No Action Alternative, it is assumed that the fuel would remain at current locations at INEEL/ANL-W, so there would be no impacts from transportation. The sodium-bonded Fast Flux Test Facility driver spent nuclear fuels and other miscellaneous fuels are assumed to be at, or brought to, Idaho, consistent with the Record of Decision for the Programmatic Spent Nuclear Fuel EIS, as amended (61 FR 9441).

4.9.2 Onsite Transportation Impacts - Alternatives 1, 2, 4 and 6

Approximately 17 shipments with the model TN-FSV cask or 43 shipments with the model NAC-LWT cask would be made by DOE under all alternatives. The analysis assumes that 43 shipments are made. The total distance traveled by trucks carrying radioactive materials on public roads located on the INEEL site would be 1,660 kilometers (1,000 miles). The dose rate from the cask is conservatively estimated to equal the maximum regulatory limit of 10 millirem per hour at two meters from the vehicle.

Impacts of Onsite Incident-Free Transportation

The dose to transportation workers from all transportation activities required by these alternatives has been estimated at 4.7×10^{-5} person-rem; the dose to the public would be 3.5×10^{-4} person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.9×10^{-8} latent cancer fatalities among transportation workers and 1.7×10^{-7} latent cancer fatalities in the total affected population over the duration of the transportation activities. Latent cancer fatalities resulting from radiological exposures were estimated by multiplying the occupational (worker) dose by 4×10^{-4} latent cancer fatalities per person-rem of exposure, and the public accident and accident-free doses by 5×10^{-4} latent cancer fatalities per person-rem of exposure (ICRP 1991).

Impacts of Onsite Accidents During Ground Transportation

The maximum foreseeable onsite transportation accident under these alternatives (occurrence probability would be more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents was estimated to be lower than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under these alternative are as follows: a radiological dose to the population of less than 1×10^{-12} person-rem, resulting in less than 1×10^{-15} latent cancer fatalities; and traffic accidents resulting in 8.2×10^{-7} traffic fatalities.

4.9.3 On- and Offsite Transportation Impacts - Alternatives 3 and 5

In addition to the onsite transportation described above, Alternatives 3 and 5 require shipment of declad and cleaned EBR-II and Fermi-1 blanket spent nuclear fuel from ANL-W to SRS. The impacts for these alternatives include both on- and offsite transportation. The total distance traveled on public roads by trucks carrying radioactive materials (packaged declad and cleaned blanket spent nuclear fuels) would be 110,700 kilometers (69,000 miles). The dose rate from the cask traveling to SRS is calculated from the contents and the shielding capabilities of the cask.

Impacts of On- and Offsite Incident-Free Transportation

The dose to transportation workers from all transportation activities required by these alternatives has been estimated at 2.0×10^{-3} person-rem; the dose to the public would be 1.3×10^{-2} person-rem. Accordingly, incident-free transportation of radioactive material would result in 7.9×10^{-7} latent cancer fatalities among transportation workers and 6.1×10^{-6} latent cancer fatalities 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 2.0×10^{-4}.

Impacts of On- and Offsite Accidents During Ground Transportation

The maximum foreseeable offsite transportation accident under these alternatives (occurrence probability would be more than 1×10^{-7} per year) is shipment of blanket spent nuclear fuels from DOE's facility at ANL-W to SRS, with a severity category V accident[1] in a suburban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.46 person-rem to the public, with an associated 2.3×10^{-4} latent cancer fatalities, and a dose of 1.9×10^{-3} to the hypothetical maximally exposed individual with a latent cancer fatality risk of 9.9×10^{-7}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of accident, or occurrence in a more densely populated area was estimated to have a probability of lower than 1.0×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a radiological dose to the population of 3.0×10^{-6} person-rem would result in 1.5×10^{-9} latent cancer fatalities; and traffic accidents would result in 0.002 traffic fatalities.

4.10 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 actions associated with the maximum impacts for the treatment and management of sodium-bonded spent nuclear fuel at ANL-W and SRS; other actions associated with onsite activities; and offsite activities with the potential for related environmental impacts. Although it is unlikely that the alternative with the maximum impact would be implemented to treat and manage 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).

[1] *A category V accident is defined as a hypothetical accident in which the strain on the inner shell of the Type B truck cask, caused by an impact, is between two percent and 30 percent of the maximum, and the centerline of the lead shielding is heated to 343 °C, or less, by a fire.*

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, and (5) waste generation. Discussion of cumulative impacts for the following resources are omitted because impacts from the proposed treatment and management of sodium-bonded spent nuclear fuel would be so small that their potential contribution to cumulative impacts would be negligible: land resources, site infrastructure, geologic resources, ecological resources, and cultural and paleontological resources.

For determining the impact to air, water, socioeconomic, human health, 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.

4.10.1 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 this assessment (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 interactions of plant emissions and no major cumulative impact on air or water use. Reasonably foreseeable offsite actions evaluated in this EIS are presented in **Table 4–62**.

Table 4–62 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	FMC phosphate manufacturing plant to reduce number of furnaces from 4 to 3 within the next two years; 25-30 jobs could be lost.
Target Department Store, Idaho Falls	Target discount store and associated commercial development near the Teton Mall in Idaho Falls.
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 on the INEEL. The project would employ 40 workers.

Source: DOE 1999a.

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 ANL-W and INEEL (see Section 1.6). The NEPA document related to ANL-W and INEEL that is considered in the cumulative impacts section is the *Surplus Plutonium Disposition Draft Environmental Impact Statement* (DOE 1998d). The Surplus Plutonium Disposition EIS analyzes the activities necessary to implement DOE's disposition strategy for surplus plutonium. INEEL is considered for pit disassembly and conversion and mixed oxide fuel fabrication for plutonium disposition. If chosen, these activities would take place at INTEC. Pit disassembly and conversion would be conducted in the Fuel Processing Facility and mixed oxide fuel fabrication in a new facility. Potential impacts from these activities are included in this section.

The cumulative impacts analysis also includes the impacts from actions proposed in this 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 Chapter 4 of this EIS.

In addition, the cumulative impacts analysis accounts for other major ANL-W and INEEL operations. These major operations are presented in **Table 4–63**.

Table 4–63 Ongoing Major Projects at ANL-W and INEEL

Project Name	*Activity*
Advanced Mixed Waste Treatment Facility	Construct new facility
Auxiliary Reactor Area-II	Decontamination and decommissioning
Central Facilities Area Hot Laundry	Decontamination and decommissioning
Central Liquid Waste Processing Facility (Building 691)	Decontamination and decommissioning
Dry Fuel Storage Facility; Fuel Receiving Canning, Characterization and Shipping	Construct new facility
Electrometallurgical Process Demonstration (formerly known as Actinide Recycle Project)	Ongoing demonstration project in Fuel Conditioning Facility hot cell
Engineering Test Reactor (and Buildings 642 and 644)	Decontamination and decommissioning
Expended Core Facility Dry Cell Project	Construct new facility
Gravel Pit Expansions	Additional gravel pits as needed
Health Physics Instrument Laboratory	Contract new facility
Industrial/Commercial Landfill Expansion	Expand landfill for nonhazardous wastes
INEEL Site Operations Center	Construct new facility
Materials Test Reactor	Decontamination and decommissioning
Pit 9 Retrieval	Construct new facility
Radioactive Scrap and Waste Facility	Storage of spent nuclear fuels and radioactive scrap waste
Remote Treatment Facility	Construct new facility
Security Training Facility	Decontamination and decommissioning
Tank Farm Heel Removal Project	Construct new facility
Technology Development Center	Construct new facility
Test Area North (Buildings 620 and 656)	Decontamination and decommissioning
Test Reactor Area Filter Pits	Decontamination and decommissioning

Source: DOE 1999a, INEEL 1999.

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 and operation of facilities identified in 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.

4.10.1.1 Air Resources

It is expected that the alternatives for the treatment and management of sodium-bonded spent nuclear fuel at ANL-W will have a negligible impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Since there ostensibly would be no change from current air quality conditions at ANL-W (see Table 3–2), there would be no cumulative impacts. There also would be no contributory effect on Prevention of Significant Deterioration increment consumption at Craters of the Moon Wilderness (Class I) Area and Class II Areas.

DOE also evaluated the cumulative impacts of airborne radiological releases in terms of dose to a maximally exposed individual at the INEEL boundary and the population within 80 kilometers (50 miles) of ANL-W. The cumulative dose to the maximally exposed offsite individual would be well below the regulatory standard of 10 millirem per year (40 CFR Part 61). Summing the doses to the maximally exposed individual for the proposed action and baseline ANL-W and INEEL operations listed in Table 4–64 is an extremely conservative approach because, to get the calculated dose, the maximally exposed individual would have to occupy different physical locations at the same time, which is impossible.

The doses from current and projected activities at ANL-W and INEEL that are associated with the Advanced Mixed Waste Treatment Facility Program and this EIS would yield a cumulative dose from airborne sources (see Table 4–64).

4.10.1.2 Water Resources

There would be no liquid effluents 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 cumulative impact.

4.10.1.3 Socioeconomic Impacts

There would be no significant cumulative socioeconomic impacts from the operation of ANL-W or INEEL facilities as a result of the proposed action. Therefore, there would be no cumulative impact.

4.10.1.4 Public and Worker Health

Table 4–64 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 EISs listed in Section 1.6. In addition to estimated radiological doses to the hypothetical maximally exposed offsite individual, the offsite population, and involved workers, Table 4–64 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 0.41 millirem per year, which is well below the applicable DOE regulatory limits (10 millirem per year from the air pathway and 100 millirem per year for all pathways). The total annual population dose of 2.49 person-rem for current and projected activities translates into 0.0012 latent cancer fatalities for each year of exposure for the population living within a 80-kilometer (50-mile) radius of the ANL-W.

The annual collective dose to the involved worker population would be 576.1 person-rem. In addition, doses to individual workers would be kept 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.

Table 4–64 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to Offsite Population and Facility Workers

	Maximally exposed offsite individual		*Population* [a]		*Workers*	
Activity	*Dose (millirem)*	*Latent Cancer Fatality Risk*	*Collective Dose (person-rem)*	*Latent Cancer Fatalities*	*Collective Dose (person-rem)*	*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	1.10×10^{-9}	0.012	6.0×10^{-6}	22	0.0088
Surplus Plutonium Disposition [d]	0.016	8×10^{-9}	2.2	0.0011	345	0.14
Advanced Mixed Waste Treatment Program [e]	0.36	1.8×10^{-7}	0.048	0.000024	4.1 [f]	0.0016
Total	0.40	2.0×10^{-7}	2.5	0.0013	486.1	0.20

[a] A collective dose to the 80-kilometer (50-mile) population for atmospheric releases. There would be no liquid releases from ANL-W and INEEL facilities as a result of the proposed action.
[b] From Tables 3–7 and 3–8 of this SBSNF EIS.
[c] Alternative 6. Melt and dilute blanket and driver fuel at ANL-W.
[d] DOE 1998d: Tables 4–134 and 4–135.
[e] DOE 1999a: Tables 5.12–1 and E.4–7.
[f] Average number of workers is 50 (DOE 1999a: Table E.4–7) × 80 millirem = collective dose.

4.10.1.5 Environmental 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 cumulative impacts on public health or the environment. The implementation of any of the alternatives at ANL-W or the No Action Alternative at INEEL would result in no disproportionately high and adverse impacts on minority or low-income populations residing within potentially affected areas.

4.10.1.6 Waste Generation

As stated in the Waste Management discussions for each alternative presented earlier in Chapter 4, high-level radioactive waste, transuranic waste, low-level radioactive waste, and mixed and hazardous waste would be generated from the treatment of sodium-bonded spent nuclear fuel. The largest volume of high-level radioactive waste would be generated at ANL-W under the No Action-Direct Disposal Alternative (152 cubic meters [5,370 cubic feet]). However, as stated in the Waste Management discussions, the projected high-level radioactive waste generation rate would not require additional treatment and storage capacities beyond the current and planned INEEL capacities.

Table 4–65 lists cumulative the volumes of high-level and low-level radioactive waste, transuranic, and hazardous and mixed wastes that ANL-W and INEEL would generate. The estimated quantity of radioactive/hazardous waste from baseline operations in this forecast during the next 15 years would be 119,550 cubic meters (7.05 million cubic feet). Waste generated by Alternative 6: Melt and Dilute Blanket and Driver Fuel at ANL-W, would add a total of 1,065 cubic meters (37,610 cubic feet). During a 10-year

time period, other reasonably foreseeable activities associated with the disposition of surplus plutonium could add an additional 1,640 cubic meters (57,920 cubic feet). Therefore, the potential cumulative total amount of waste generated from ANL-W and INEEL activities would be 122,255 cubic meters (4.32 million cubic feet). It is important to note that the quantities of waste generated are not equivalent to the amounts that will require disposal. As discussed in Chapter 4 for each of the treatment alternatives, during treatment high-level radioactive material is evaporated and concentrated to a smaller volume for final disposal.

Table 4–65 Estimated Cumulative Total Waste Generation From ANL-W and INEEL Concurrent Activities (cubic meters)

Waste Type	*ANL-W and INEEL Baseline Operations* [a]	*SBSNF EIS* [b]	*Surplus Plutonium Disposition* [c]	*Total*
High-level radioactive	8,250	86	0	8,336
Low-level radioactive	64,500	924	940	66,364
Hazardous/mixed	46,800	41	60	46,901
Transuranic	0	14	640	654
Total	119,550	1,065	1,640	122,255

[a] From SBSNF EIS Table 3–9, 15 years of operation.
[b] Alternative 6. Melt and Dilute Blanket and Driver Fuel at ANL-W, 15 years of operation.
[c] DOE 1998d: Table 4–133, 10 years of operation.

The Central Facilities Area and Bonneville County landfill accepts nonhazardous and nonradioactive solid wastes 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 Cold Waste Handling Facility at INTEC is designed to inspect, recycle, shred, compact, and segregate nonhazardous waste, thereby reducing the amount of material sent to disposal.

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 wastes beyond the current and projected capacities of ANL-W and INEEL waste storage and/or management facilities.

4.10.2 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 Draft EIS (DOE 1998g), 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 and Hankook Polyester) 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 about 32 river kilometers (20 river miles) 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 1998e). 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 1999c). 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.

Surplus Plutonium Disposition Draft Environmental Impact Statement (DOE 1998d). This EIS analyzes the activities necessary to implement DOE's disposition strategy for surplus plutonium. SRS is considered the preferred location for mixed oxide fuel fabrication and plutonium immobilization facilities that would be used for plutonium disposition. SRS also is the preferred site for the pit disassembly and conversion facility. Impacts from this EIS are included in this section.

Defense Waste Processing Facility Supplemental Environmental Impact Statement (DOE 1994b). 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.

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 Chapter 4, 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 Draft EIS (DOE 1998g) and the Tritium Extraction Facility, while actions for nuclear materials, highly enriched uranium, and surplus plutonium disposition would be ongoing.

4.10.2.1 Air Resources

Table 4–66 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. The highest percentages of the regulatory standards are for sulfur dioxide concentrations for the shorter time interval (approximately 59 percent of the standard for the 24-hour averaging time); for particulate matter of less than 10 microns, 24-hour interim (approximately 54 percent of the standard); and for sulfur dioxide, 3-hour averaging time (approximately 54 percent of the standard). The remaining pollutant emissions would be below 25 percent of the applicable standards.

DOE also evaluated the cumulative impacts of airborne radiological releases in terms of dose to a maximally exposed 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 1998g) and are not included. **Table 4–67** 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 0.87 millirem per year, well below the regulatory standard of 10 millirem per year (40 CFR Part 61). Summing the doses to the maximally exposed individual for the proposed action and baseline SRS operations listed in Table 4–67 is an extremely conservative approach because, to get the calculated dose, the maximally exposed individual would have to occupy different physical locations at the same time, which is impossible.

Adding the population doses from current and projected activities at SRS, the Vogtle Plant, the SRS Spent Nuclear Fuel Management Draft EIS, and this EIS could yield a total annual cumulative dose of 39.77 person-rem from airborne sources. The total annual cumulative dose translates into 0.020 latent cancer fatalities for each year of exposure for the population living within an 80-kilometer (50-mile) radius of SRS.

Table 4–66 Estimated Maximum Cumulative Ground-Level Concentrations of Nonradiological Pollutants (micrograms per cubic meter) at SRS Boundary

Pollutant	*Averaging Time*	*Most stringent standard or guideline* [a]	*SRS baseline* [b]	*SBSNF* [c]	*Other foreseeable planned SRS activities* [d]	*Cumulative Concentration*	*Percent of Standard*
Carbon monoxide	8 hours 1 hour	10,000[e] 40,000[e]	632 5,010	1.22 9.06	20.61 89.36	653.8 5,108.4	7 13
Nitrogen dioxide	Annual	100 [e]	8.8	3.11	7.02	18.9	19
PM_{10}	Annual 24 hours (interim) 24 hours (99th percentile over 3 years)	50 [e] 150 [e] 150 [f]	4.8 80.6 (g)	less than 0.01 0.11 NA	0.05 0.29 NA	4.9 81 NA	10 54 NA
PM_{25}	3 year annual 24 hours (98th percentile over 3 years)	15 [f] 65 [f]	(g) (g)	NA NA	NA NA	NA NA	NA NA
Sulfur dioxide	Annual 24 hours 3 hours	80 [e] 365 [e] 1,300 [e]	16.3 215 690	less than 0.01 0.12 0.91	0.14 1.63 5.38	16.5 216.8 696.3	21 59 54

N/A = Not available.
[a] The more stringent Federal and state standards are presented if both exist for the averaging period.
[b] Data from Table 3–13 of this EIS.
[c] Alternative 3, PUREX Process Blanket Fuel at SRS F-Canyon.
[d] Data compiled from SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g: Table 5–1).
[e] Federal and state standard.
[f] Federal standard.
[g] No data available with which to assess particulate matter concentrations.
Key: PM_{10}= Particulate matter less than or equal to *n* microns.

4.10.2.2 Water 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 effluents to surface water would be PUREX processing. The major sources of liquid effluents 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 Creek. 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 1998g).

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 Draft EIS (DOE 1998g), 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 Draft 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 4–67 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Maximally Exposed Offsite Individual and Population in the 80-Kilometer (50-Mile) Radius From Airborne Releases at SRS

	Offsite Population			
	Maximally Exposed Offsite Individual		*Population Within 80 Kilometers (50 Miles)*	
Activity	*Dose (millirem)*	*Latent Cancer Fatality Risk*	*Collective Dose (person-rem)*	*Latent Cancer Fatalities*
SRS Baseline[a]	0.050	2.5×10^{-8}	5.5	2.8×10^{-3}
This SBSNF EIS[b]	0.00039	2.0×10^{-10}	0.019	9.5×10^{-6}
Management of Spent Nuclear Fuel[c]	0.015	7.5×10^{-9}	0.56	0.00028
Surplus Highly Enriched Uranium Disposition[c]	0.0025	1.3×10^{-9}	0.16	0.00008
Interim Management of Nuclear Materials[c]	0.77	3.9×10^{-7}	31	0.016
Tritium Extraction Facility[c]	0.02	1.0×10^{-8}	0.77	0.00039
Surplus Plutonium Disposition[c]	0.004	2.0×10^{-9}	1.6	0.0008
Management of Plutonium Residues/ Scrub Alloy[c]	0.003	1.5×10^{-9}	0.038	0.000019
Defense Waste Processing Facility [c]	0.001	5.0×10^{-10}	0.071	0.000036
DOE complex miscellaneous components [c]	0.0044	2.2×10^{-9}	0.007	3.3×10^{-6}
Vogtle Plant [c]	0.00054	2.7×10^{-10}	0.042	0.000021
Total	0.87	4.4×10^{-7}	39.8	0.020

[a] Data from Table 3–16 of this SBSNF EIS.
[b] Alternative 3. PUREX Process Blanket Fuel at SRS F-Canyon.
[c] Data from SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g: Table 5–2 maximum-impact alternative).
Key: HEU = Highly enriched uranium.

Table 4–68 summarizes the estimated cumulative radiological doses from waterborne sources to human receptors downstream from SRS. Liquid effluents 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 0.26 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 3.24 person-rem from liquid sources. This translates into 0.0016 latent cancer fatalities for each year of exposure of the population living within an 80-kilometer (50-mile) radius of SRS.

Table 4–68 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to Offsite Population in the 80-Kilometer (50-Mile) Radius From Liquid Releases at SRS

	Offsite Population			
	Maximally Exposed Offsite Individual		*Population Within 80 Kilometer (50-miles)*	
Activity	*Dose (millirem)*	*Latent Cancer Fatality Risk*	*Collective Dose (person-rem)*	*Latent Cancer Fatalities*
SRS Baseline[a]	0.13	6.5×10^{-8}	2.4	0.0012
SBSNF EIS [b]	0.00012	6.0×10^{-11}	0.00068	3.4×10^{-7}
Management of Spent Nuclear Fuel [c]	0.057	2.9×10^{-8}	0.19	0.000095
Surplus Highly Enriched Uranium Disposition [c]	(d)	(d)	(d)	(d)
Interim Mgmt of Nuclear Materials [c]	0.022	1.1×10^{-8}	0.65	0.00033
Tritium Extraction Facility [c]	(d)	(d)	(d)	(d)
Defense Waste Processing Facility [c]	(d)	(d)	(d)	(d)
Surplus Plutonium Disposition [c]	(d)	(d)	(d)	(d)
Management Plutonium Residues/ Scrub Alloy [c]	(d)	(d)	(d)	(d)
DOE complex miscellaneous components [c]	0.000042	2.1×10^{-11}	0.00024	1.2×10^{-7}
Sodium-Bonded Spent Nuclear Fuel [c]	0.000042	2.1×10^{-11}	0.00024	1.2×10^{-7}
Plant Vogtle [c]	0.054	2.7×10^{-8}	0.0025	1.3×10^{-6}
Total	0.26	1.3×10^{-7}	3.24	0.0016

[a] Data from Table 3–16 of this SBSNF EIS.
[b] Alternative 3. PUREX Process Blanket Fuel at SRS F-Canyon.
[c] Data from SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g: Table 5-3 maximum-impact alternative).
[d] Less than minimum reportable levels.
Key: HEU = Highly enriched uranium.

4.10.2.3 Socioeconomic Impacts

Cumulative regional economic and population changes from construction and operation of the Transfer and Storage Facility or the Transfer, Storage, and Treatment Facility, both of which are described in the SRS Spent Nuclear Fuel Management Draft EIS, consider the impacts of other coincident economic development projects such as Bridgestone-Firestone and Hankook Synthetics.

Bridgestone-Firestone is building a $435 million tire manufacturing plant in Aiken County, South Carolina, that will employ 800 workers. Hankook Synthetics announced plans to build an $850 million polyester fiber plant in Richmond County, Georgia, that will employ 500 workers. Both the Bridgestone-Firestone and Hankook projects are expected to complete construction and be in operation by 2000. Thus, these two projects should not impact the construction workforce for the Transfer and Storage Facility or the Transfer, Storage, and Treatment Facility, which are not scheduled to be constructed until 2000. Competition for construction workers should not overlap (DOE 1998g).

Construction of the Transfer and Storage Facility or the transfer and storage phase of the Transfer, Storage, and Treatment Facility, both of which are described in the SRS Spent Nuclear Fuel Management Draft EIS, would start in 2000, employ 500 workers (375 construction and 125 professional), and require two years to complete. The treatment phase would begin construction at the completion of the transfer and storage phases and also could employ as many as 500 workers and take as long as two years to complete. No additional workers would be required during operations since existing SRS employees would assume those positions. There would be no significant cumulative socioeconomic impacts from construction or operation of the Transfer and Storage Facility or the Transfer, Storage, and Treatment Facility (DOE 1998g).

4.10.2.4 Public and Worker Health

Table 4–69 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 involved workers, Table 4–69 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 1.13 millirem per year, which is well below the applicable DOE regulatory limits (10 millirem per year from the air pathway; four 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 43.07 person-rem translates into 0.02 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 the involved worker population would be 1,152 person-rem. In addition, doses to individual workers would be kept 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.

4.10.2.5 Environmental 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 cumulative impacts on public health or the environment. 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.

4.10.2.6 Waste Generation

As stated in Sections 4.5.6 and 4.7.6, low-level and high-level radioactive waste, transuranic waste, mixed waste, and hazardous waste would be generated from the treatment of sodium-bonded spent nuclear fuel. The largest volume of high-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 high-level radioactive and transuranic waste generation rates would not require additional treatment and storage capacities beyond the current and planned SRS capacities.

Table 4–69 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to Offsite Population and Facility Workers

	Maximally Exposed Offsite Individual				*Population*[a]				*Workers*	
Activity	*Dose from Airborne Releases (millirem)*	*Dose from Liquid Releases (millirem)*	*Total Dose (millirem)*	*Latent Cancer Fatalities Risk*	*Collective Dose from Airborne Releases (person-rem)*	*Collective Dose from Liquid Releases (person-rem)*	*Total Collective Dose*	*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
Surplus Highly Enriched Uranium Disposition[d]	0.0025	(e)	0.0025	1.3×10^{-8}	0.16	(e)	0.16	0.00008	11	0.00044
Interim Management of Nuclear Materials[d]	0.77	0.022	0.79	4.0×10^{-7}	31	0.65	31.7	0.016	130	0.052
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
Surplus Plutonium Disposition [d]	0.004	(e)	0.004	2.0×10^{-9}	1.6	(e)	1.6	0.0008	541	0.22
Management Plutonium Residues/ Scrub Alloy [d]	0.003	(e)	0.003	1.5×10^{-9}	0.038	(e)	0.038	0.000019	47	0.019
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

Activity	Maximally Exposed Offsite Individual				Population[a]				Workers	
	Dose from Airborne Releases (millirem)	Dose from Liquid Releases (millirem)	Total Dose (millirem)	Latent Cancer Fatalities Risk	Collective Dose from Airborne Releases (person-rem)	Collective Dose from Liquid Releases (person-rem)	Total Collective Dose	Latent Cancer Fatalities	Collective Dose (person-rem)	Excess Latent cancer fatalities
Sodium-Bonded Spent Nuclear Fuel [d]	0.00012	0.000042	0.00016	8.1×10^{-11}	0.0042	0.00024	0.004	2.2×10^{-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	NA	NA
Total	0.87	0.26	1.13	5.9×10^{-7}	39.77	3.24	43.07	0.022	1,115	0.45

[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 3–16 and 3–17 of this EIS.
[c] Alternative 3: PUREX Process Blanket Fuel at SRS F-Canyon.
[d] Data from SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g: Table 5-4, maximum-impact alternative).
[e] Less than minimum reportable levels.

Table 4–70 lists the cumulative volumes of high- and low-level radioactive, transuranic, hazardous, and mixed wastes that SRS would generate. The table includes data from the SRS 30-year expected waste forecast (DOE 1998g). 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 and the Interim Management of Nuclear Materials at SRS EIS; (3) some investigation-derived wastes are handled as hazardous waste per Resource Conservation and Recovery Act (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 1998g).

Table 4–70 Estimated Cumulative Total Waste Generation from SRS Concurrent Activities (Cubic Meters)

Waste Type	*SRS Baseline Operations*[a]	*SBSNF EIS* [b]	*Management*[a] *of SNF*	*ER/D&D*[a]	*Other Waste Volume*[a]	*Total*
High-level radioactive	150,750	106	11,000	0	69,642	231,498
Low-level radioactive	343,710	1,398	140,000	132,000	194,553	811,661
Hazardous/mixed	90,450	3	270	575,180	5,156	671,059
Transuranic	18,090	17	3,700	4,820	8,760	35,387
Total	603,000	1,524	154,970	712,000	278,111	1,749,605

[a] Data from SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g) maximum-impact alternative, Table 5-5, based on a total of 30-year expected waste forecast, which includes previously generated waste.
[b] Alternative 5. Melt and Dilute Blanket Fuel at SRS Building 105-L, 15-years of operation.
Key: ER/D&D = environmental restoration/decontamination & deactivation; based on a total 30-year expected waste forecast.

The estimated quantity of radioactive/hazardous waste from SRS operations in this forecast during the next 30 years would be 603,000 cubic meters (21.3 million cubic feet). Waste generated by Alternative 5: Melt and Dilute Blanket Fuel at SRS Building 105-L, would add a total of 1,524 cubic meters (53,820 cubic feet). Waste generated from the conventional (PUREX) processing option described in the SRS Spent Nuclear Fuel Management Draft 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 deactivation activities would have a 30-year expected forecast of 712,000 cubic meters (25.1 million cubic feet) (DOE 1998g). During this same time period, other reasonably foreseeable activities that were not included in the 30-year forecast would add an additional 278,111 cubic meters (9.82 million cubic feet). Therefore, the potential cumulative amount of waste generated from SRS activities during the period of interest would be 1,749,605 cubic meters (61.8 million cubic feet). It is important to note that the quantities of waste generated are not equivalent to the amounts that will require disposal. As discussed in Chapter 4 for each of the treatment alternatives at SRS, during treatment 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 wastes 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 wastes while reducing the environmental consequences associated with construction and operation of multiple county-level facilities (DOE 1998g). 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 1998g). 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 wastes. 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 wastes beyond the current and projected capacities of SRS waste storage and/or management facilities.

4.11 PROGRAMMATIC CONSIDERATIONS

The programmatic considerations presented below is a programmative perspective of the alternatives vis-a-vis the current regulatory environment regarding spent nuclear fuel and high-level radioactive waste and the expected time frame for the disposal of DOE-owned spent nuclear fuel or high-level radioactive waste in a geologic repository.

4.11.1 Regulatory Environment Considerations

Prior to the acceptance of spent nuclear fuel or high-level radioactive waste at the proposed repository, certain regulatory 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 March 19, 1999, draft *Waste Acceptance System Requirements Document* (DOE/RW) (DOE 1999b).

One of the key non-DOE (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 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 the repository acceptance criteria. Under all other 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 part 268, subpart D." Deactivation is the waste treatment technology 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 wastes that exhibit 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 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 still be considered to be reactive and ignitable, and therefore, it may not be accepted for disposal at the potential 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 described as reactive, and in some cases pyrophoric; however, it would not be a RCRA hazardous characteristic 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 the Civilian Radioactive Waste Management Office's specific acceptance criteria for DOE's spent nuclear fuel and high-level radioactive waste. For high-level radioactive waste, the document specifies a standard vitrified high-level radioactive waste form as borosilicate glass. Specific acceptance criteria standards have not been developed for other treated waste forms of high-level radioactive waste (e.g., ceramic forms and metal waste forms). 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. However, specific acceptance criteria have not been developed for spent nuclear fuel that has been melted into a liquid form and then solidified (e.g., conditioned). The No Action Alternative may be able to meet this requirement for the disposal canisters; however, it may not meet all of the other requirements previously discussed (e.g., reactivity, ignitability, and RCRA regulations); therefore, it may not be acceptable for disposal.

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 facilities 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 metal 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 facilities license.

For Alternative 2 (blanket fuel), the specific acceptance criteria for canistered spent nuclear fuel would apply and most likely could 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.

4.11.2 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 *Viability Assessment of a Repository at Yucca Mountain*, Volume 1, December 1998, DOE/RW (DOE 1998h). 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 1998h) 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 indicates 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. A delayed decision for 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 for completion by 2009. Delaying a decision until after 2010 would delay the completion 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 will 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 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. A delayed decision until after 2010 would push completion to approximately 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 EIS (DOE 1998g). 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 be completed by 2009, well before processing begins. Delaying a decision until 2010 would push the completion of the decladding activities to 2019, which could 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 2005 and be completed by 2015. Delaying a decision until 2010 would push completion to approximately 2025.

Table 4–71 provides a summary of 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 4–71 Year of Completing Treatment[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.

4.12 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.

4.13 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.

4.13.1 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 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 radioactive waste, which would either be treated and stored on site at ANL-W or SRS, or transported and managed off site at low-level radioactive 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 could 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 could be affected by increases in various chemical or radiological constituents in the routine effluent releases. 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.

4.13.2 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 wastes. 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.

4.13.3 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. Operation of any proposed facility would generate nonrecyclable waste streams, such as radiological and nonradiological solid wastes 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 wastes 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.

4.14 REFERENCES

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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 Department of Energy environmental, safety and health laws, regulations, and orders 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 to ensure protection of public health, safety; and the environment through compliance with all applicable Federal and state laws, regulations, orders, and other requirements. This section describes the environmental, safety, and health laws; regulations; and orders that apply to the proposed action and alternatives.

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 environmental impact statement for any major Federal action that can have a significant environmental impact.

Applicable implementing regulations for NEPA include the Council on Environmental Quality Implementing Regulations (40 CFR 1500 *et seq*.) and DOE 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 are generally 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 provides for research, development, and demonstration activities regarding disposal of high-level radioactive waste and spent nuclear fuel that does not result from defense activities. 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 Nuclear Waste Policy Act also established the Office of Civilian Radioactive Waste Management (OCRWM, 42 U.S.C. 10224) and the Nuclear Waste Fund (42 U.S.C. 10222). The Energy Policy Act of 1992, Section 801, directed the Environmental Agency 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 wastes; 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 the 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 Facility Compliance Act of 1992 (42 U.S.C. 6961 *et seq*.)—Section 102(a)(3) of the Federal Facility 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.

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.

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.

Air emissions are regulated by the EPA under 40 CFR Parts 50 through 99. Radionuclide emissions are regulated under the National Emission Standards for Hazardous Air Pollutants Program under 40 CFR Part 61.

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 discharge effluents are altered.

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.

Hazardous Material Transportation Act of 1975 (49 U.S.C. 5105 *et seq*.)—The Hazardous Material Transportation Act requires the 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 Nuclear Regulatory Commission (under the Atomic Energy Act) and with the EPA (under RCRA) when covering the same activities.

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 properly identified and appropriate mitigative actions are implemented.

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 Department of 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.

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 the OSHA and the EPA both have a mandate to reduce exposures to toxic substances, the 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. The 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 the 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, Environmental Protection Agency 33/50 Pollution Prevention Program. The goal for facilities involved in compliance with Section 313 is 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 requires 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.

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 and 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.

5.1.3 DOE Orders

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 Orders and regulations, an extensive system of standards and requirements 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 Environmental Impact Statement (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 Orders are issued in support of health, safety, and environmental programs. Many of the DOE Orders have been revised and reorganized to reduce duplication and eliminate obsolete provisions. The new DOE Order organization is by series, with each number identified by three digits, and is intended to include all DOE policies, manuals, requirement documents, notices, guides, and orders. The remaining DOE Orders, which are identified by four digits, are expected to be revised and converted to the new DOE numbering system over the next two years. The major DOE Orders pertaining to the proposed action and alternatives are listed in **Table 5–1**.

Table 5–1 Relevant DOE Orders (as of June 1999)

DOE Order	*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 414.1	Quality Assurance (11/24/98)
O 420.1	Facility Safety (10/13/95; Change 2, 10/24/96)
O 440.1A	Worker Protection Management for DOE Federal and Contractor Employees (03/27/98)
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)

DOE Order	Subject
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)
5633.3B	Control and Accountability of Nuclear Materials (09/07/94)
5660.1B	Management of Nuclear Materials (05/26/94)
Energy Research and Technology	
5820.2A	Radioactive Waste Management (09/26/88)
Design	
6430.1A	General Design Criteria (04/06/89)

5.1.4 State 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**.

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 and 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.
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.

Law/Regulation/Agreement	*Citation*	*Potential Requirements*
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 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 wastes.
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.

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 U.S. Nuclear Regulatory Commission. 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 Nuclear Regulatory Commission 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 U.S. Nuclear Regulatory Commission 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 U.S. Nuclear Regulatory Commission 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).

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 clearly identified 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 U.S. Nuclear Regulatory Commission.

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 emissions 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 (for example 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 enables employees to be aware of the dangers they face from hazardous materials at their workplace.

Emergency Management and Assistance (44 CFR 1.1)—This regulation contains the policies and procedures for the Federal Emergency Management Act, National Flood Insurance Program, Federal Crime Insurance Program, Fire Prevention and Control Program, Disaster Assistance Program, and Preparedness Program, including radiological planning and preparedness.

Hazardous Materials Tables and Communications, Emergency Response Information Requirements (49 CFR Part 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 has initiated the consultations at the two sites and will report the status of these consultations in the Final EIS for the treatment and management of sodium-bonded spent nuclear fuel.

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 — Geographic subdivisions 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 Wastes — Wastes 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 (ALARA) — 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.

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 — Glass typically containing 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, its purpose is 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 fuel elements usually made of aluminum, stainless steel, or zirconium alloy, 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.

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.

Contact-handled Waste — Packaged waste whose external surface dose rates does not exceed 200 millirem per hour.

Container — With regard to radioactive wastes, 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.

^{o}C (degrees Celcius) — 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.

^{o}F (degrees Farhenheit) — A unit for measuring temperature using the Farhenheit 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 it to another material.

Disposal — The isolation of radioactive wastes 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, these fuels are called drivers.

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 — A 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 wastes 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 EIS, scheduled for January 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, will not have a significant effect on the human environment and will 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 — Nuclei formed by the fission of heavy elements (primary fission products); also, the nuclei formed by the decay of the primary fission products, many of which are radioactive.

Fissionable Material — Material that could undergo fission by the absorption of fast neutrons.

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 per second^2 (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.

(1) Effects on genetic material in reproductive cells cause trait modifications that can be passed from parents to offspring.

(2) Effects on genetic material in nonreproductive cells 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 (a) the geologic repository operations area, and (b) 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 (HEPA) — A filter used to remove very small particulates from dry gaseous effluent streams.

High-Level Radioactive Waste — The 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 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 Fatalities — Fatalities associated with acute and chronic environmental exposures to chemical or radiation that occur within 30 years of 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 DOE 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 (MTHM) — 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 RCRA, 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 effluents 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 chemical solution (e.g., a boron or component sheet 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 NO (nitrogen oxide) and NO_2 (nitrogen dioxide). These are produced in the combustion of fossil fuels 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 that are highly resistant to oxidation and corrosion, such as zirconium, niobium, and gold.

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.

***Occupational Safety and Health Administration* —** Oversees and regulates workplace health and safety, created by the Occupational Safety and Health Act of 1970.

***Off-gas* —** Volatile and semi-volatile gaseous products that are 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 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. Naturally occurring radiation is indistinguishable from induced radiation.

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, in order to recycle such materials primarily for defense programs. Historically, reprocessing has involved aqueous chemical separations 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 safety 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 — Wastes 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 which 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 standardized DOE canister which is a stainless steel, right circular cylinder with a nominal outside diameter of 45.7 centimeters (18 inches), a nominal thickness of .59 centimeters (.375 inches) and a maximum overall length of 3 meters (118.11 inches) with a usable length of 2.55 meters (100.28 inches). The standard 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. It is not a mixed waste. (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 1 - applied to wilderness areas, wild and scenic rivers, and other similar situations; Class 2 - contrasts are seen, but do not attract attention; Class 3 - contrasts caused by a cultural activity are evident, but remain subordinate to the existing landscape; Class 4 - contrasts that attract attention and are dominant features of the landscape in terms of scale, but repeat the contrast of the characteristic landscape; Class 5 - applied to areas where unacceptable cultural modification has lowered scenic quality (where the natural character of the landscape has been disturbed to a point where rehabilitation is needed to bring it up to one of the four other classifications).

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, often halogenated, 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 — Any group 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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Approximately 273 copies of the Draft EIS were sent to stakeholders
Approximately 1,560 copies of the Summary of the Draft EIS were sent to stakeholders

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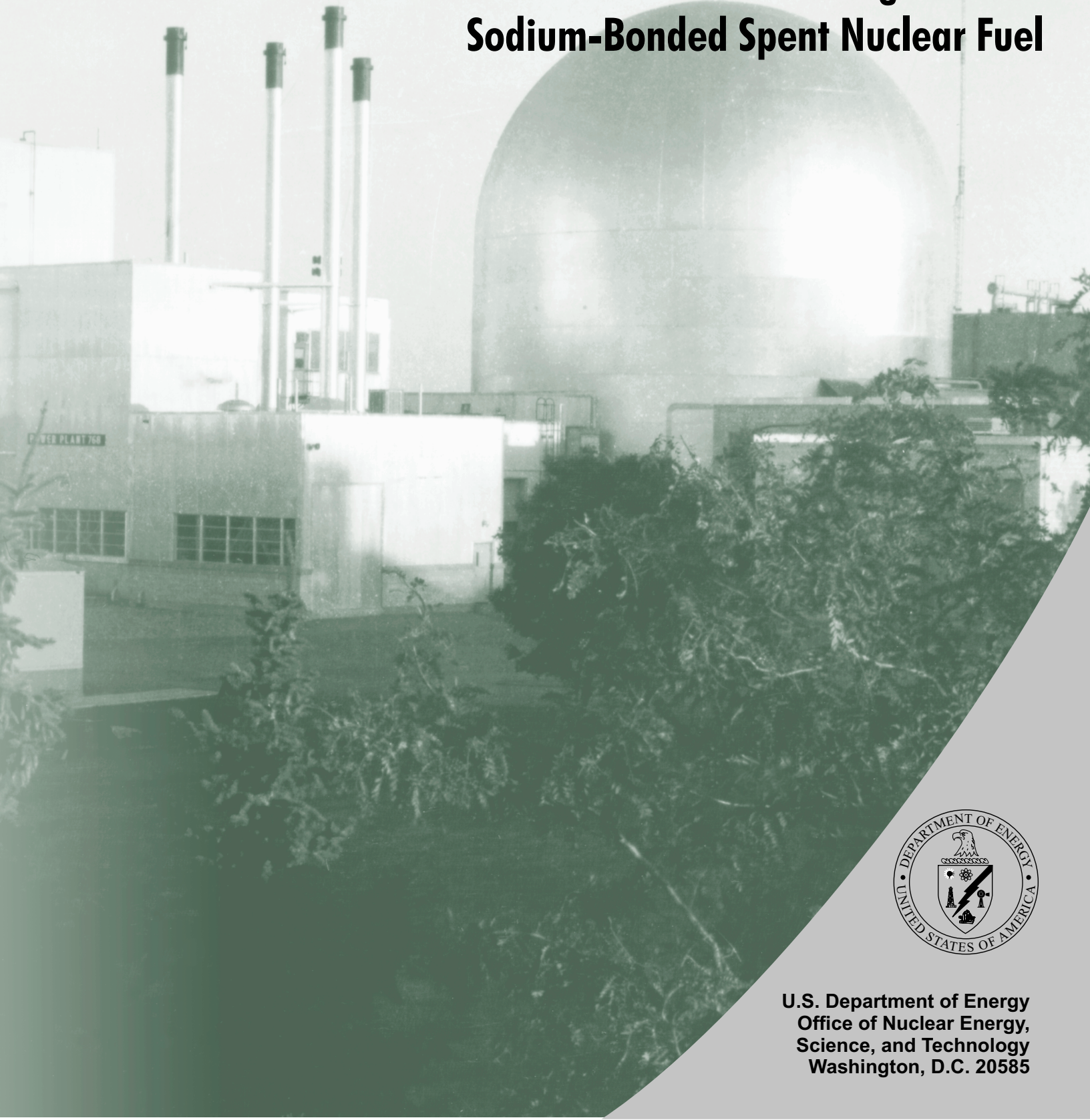
for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel



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Abstract: The Department of Energy (DOE) is responsible for the safe and efficient management of several different types of spent nuclear fuel. One type of spent nuclear fuel that may not be suitable for disposal in a geologic repository without treatment is the DOE-owned sodium-bonded spent nuclear fuel. Sodium-bonded spent nuclear 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 material could complicate the process of qualifying and licensing such spent nuclear fuel for disposal in a geologic repository. Currently, more than 98 percent of DOE's sodium-bonded spent nuclear fuel is located at the Idaho National Engineering and Environmental Laboratory (INEEL). In a 1995 agreement with the State of Idaho, DOE committed to remove all spent nuclear fuel from Idaho by 2035.

Several technologies for spent nuclear fuel treatment are under development and might facilitate qualification and licensing for ultimate disposal. The most developed technology is the electrometallurgical treatment of sodium-bonded spent nuclear fuel at Argonne National Laboratory-West (ANL-W). This EIS evaluates the potential environmental impacts associated with the treatment of sodium-bonded spent nuclear fuel in one or more spent nuclear fuel management facilities: ANL-W at INEEL (near Idaho Falls, Idaho) and either the F-Canyon or Building 105-L at the Savannah River Site (near Aiken, South Carolina). The EIS analyzes under the proposed action the electrometallurgical process, the plutonium-uranium extraction (PUREX) process, direct disposal in high-integrity cans with the sodium removed, and the melt and dilute process. The EIS also evaluates the continued storage of sodium-bonded spent nuclear fuel and direct disposal without treatment under the No Action Alternative.

Public Comments: In preparing this Draft EIS, DOE considered comments received from the public during the scoping process (February 22, 1999 to April 8, 1999). Comments on this Draft EIS may be submitted during the 45-day comment period. Public meetings on this EIS will also be held during the comment period. The dates, times, and locations of these meetings will be announced shortly after issuance of this Draft EIS.

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ACRONYMS, ABBREVIATIONS, AND CONVERSION CHARTS

ANL	Argonne National Laboratory
ANL-W	Argonne National Laboratory-West
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
EMT	Electrometallurgical Treatment (of spent fuel)
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
IAEA	International Atomic Energy Agency
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
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
P.L.	Public Law
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

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 5/9ths	Celsius	Celsius	Multiply by 9/5ths, 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}

APPENDIX A
THE PUBLIC SCOPING PROCESS

A.1 SCOPING PROCESS DESCRIPTION

As a preliminary step in the development of an environmental impact statement (EIS), regulations established by the Council on Environmental Quality (40 CFR 1501.7) and the U.S. Department of Energy (DOE) require "an early and open process for determining the scope of issues to be addressed and for identifying the significant issues related to a proposed action." The purpose of this scoping process is: (1) to inform the public about a proposed action and the alternatives being considered and (2) to identify and/or clarify those issues considered most relevant by the public.

On February 22, 1999, DOE published in the *Federal Register* a Notice of Intent to prepare an EIS for the treatment of sodium-bonded spent nuclear fuel (SBSNF EIS). As shown in **Figure A–1**, the scoping process is one of the opportunities for public involvement required as part of the National Environmental Policy Act (NEPA) process. The Notice of Intent listed the alternatives and issues initially identified by DOE for evaluation in the EIS. Members of the public, civic leaders, and other interested parties were invited to comment on these issues and to suggest additional issues that should be considered in the EIS. The Notice of Intent also informed the public that comments on the proposed action could be communicated via U.S. mail, a special DOE web site on the Internet, a toll-free phone line, a toll-free fax line, or in person at one of four public meetings.

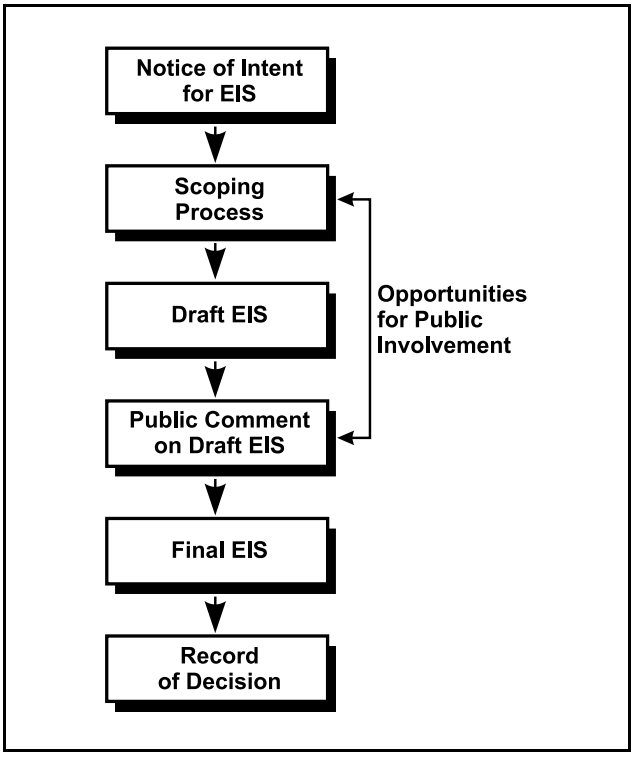

Figure A–1 NEPA Process

Four public scoping meetings were held at locations in Idaho, South Carolina, and Virginia, near the Washington, DC, metropolitan area. The first public meeting was attended by about 60 members of the public 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 about 7 members of the public. Approximately 10 members of the public 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 about 8 members of the public.

As a result of previous experience and positive responses from attendees of other DOE/NEPA public meetings and hearings, DOE chose an interactive format for the scoping meetings. Each meeting began with a presentation by a DOE representative who explained the proposed action. Afterwards, an impartial facilitator opened the floor to questions, comments, and concerns from the audience. DOE and national laboratory personnel were available to respond to the questions and comments as needed. A court reporter was provided at each of the meetings to record the oral comments, and personnel were available to receive any written statements or comments that were submitted at the meetings. In addition, the public was encouraged to submit written or verbal comments via letters, the DOE Internet web site, the toll-free phone line, or the toll-free fax line until the end of the scoping period on April 8, 1999 (45 days after publication of the Notice of Intent).

It should be noted that, for EIS public scoping purposes, a comment is defined as a single statement or opinion concerning a specific issue. Any statement may contain many separate comments. Most of the verbal and written public statements submitted during the EIS scoping period contained multiple comments on various individual issues.

A.2 SCOPING PROCESS RESULTS

Approximately 228 comments were received from citizens, interested groups, and other stakeholders during the public scoping comment period. Of these, 109 were verbal comments made during the public meetings. The remainder of the comments (119) either were submitted at the public meetings in written form or were received via mail, Internet, fax, or phone during the scoping comment period. In cases where a single commentor provided similar or identical comments both orally at the public meetings and in writing, each individual comment was counted once (i.e., repetitions were not counted).

Many members of the public who spoke at the public meetings asked specific, technical questions about the proposed action that were answered by the DOE and national laboratory representatives at each meeting. Primary areas of interest included:

- Waste volume reduction
- Nature of the spent nuclear fuel wastes at Argonne National Laboratory-West (ANL-W)
- Waste forms characterization
- Waste disposition and qualification (repository acceptance criteria)
- Plutonium-uranium extraction (PUREX)
- Use of facilities
- Nonproliferation impacts
- Transportation
- Demonstration project

The comments obtained through the overall public scoping process addressed several key issues. A number of persons commented on the schedule for the EIS. Many said the Draft EIS should not be issued for public comment before publication of other reports, such as a waste qualification assessment from the National Research Council; the National Academy of Science's Independent Assessment Final Report on the demonstration project; a nonproliferation assessment report by the DOE Office of Nonproliferation and National Security; and an independent study of the costs of the proposed action. Several commentors also said 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 the public should have an opportunity to comment on DOE's ongoing independent nonproliferation assessment within the same time frame as the Draft EIS, or that this EIS should be delayed until the nonproliferation assessment is publicly

available. Some suggested the nonproliferation assessment be included in the EIS. A few commentors expressed the opinion that electrometallurgical treatment of spent nuclear fuels is a proliferation-prone technology.

Waste was another issue that was frequently cited. 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 alternatives 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 the geologic repository.

Regarding the alternative technologies being evaluated as part of this EIS, the commentors generally agreed that DOE should evaluate in detail all of the alternative 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 fuels 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. A spokesman for the Shoshone-Bannock Tribe, which considers the Idaho National Engineering and Environmental Laboratory (INEEL) land to be part of their original territory, expressed confidence that the proposed electrometallurgical treatment process would not impact the land's cultural resources or native species. Other commentors wanted the EIS to explain whether there were any environmental threats associated with continued storage of the spent nuclear fuel in Idaho and the nature of the environmental impacts of all the alternative technologies listed in the Notice of Intent. Transportation-related comments were rare, but reflected some public concern about the safety and security of transporting spent nuclear fuel and other waste products over long distances.

Some commentors simply opposed the proposed action as a waste of money or an example of corporate welfare. Others stated that DOE already has determined its choice of alternatives and is merely engaging in a show process that meets the bare minimum legal requirements.

A.3 COMMENT DISPOSITION AND ISSUE IDENTIFICATION

Comments received during the scoping period were systematically reviewed and evaluated to determine whether the issues raised fell within or outside the scope of the EIS as contemplated in the Notice of Intent (64 FR 8553). Where possible, comments on similar or related topics were grouped under comment categories as a means of summarizing the comments. An attempt was made to avoid duplication in counting the number of comments received; however, comments submitted in both written and verbal form may have been counted twice in some cases. The comment categories were used to identify specific issues of public concern. After the issues were identified, they were evaluated to determine whether they fell within or outside the scope of the EIS. Some issues were found to be already "in scope," i.e., they were among the EIS issues already

identified by DOE for inclusion in the EIS. **Table A–1** lists these issues along with references to the specific EIS sections where each issue is discussed.

Table A–1 Issues Already Included in the EIS (In Scope)

Issues	*No. of Comments*	*EIS References*
The EIS should specify what the stable sodium compound technology alternative is and how it is derived	1	Section 2.3
The EIS should explain how the PUREX process, which could not handle sodium-bonded spent nuclear fuel before [in the aluminum-bonded Spent Nuclear Fuel EIS], now is considered an acceptable alternative for the proposed action.	1	Section 2.3.2
DOE says the Savannah River PUREX process will handle the sodium, but more research will be needed to improve the sodium-handling ability of the PUREX process. If research is needed to make the Savannah River PUREX process work for sodium, DOE might as well do research in Idaho in some different process. I'm in favor of Idaho; DOE should be cautious about talking PUREX and sodium-bonded stuff.	2	Section 2.3.2
The EIS should evaluate whether changes in the PUREX process would be needed to accommodate this material. After the plutonium is separated in the PUREX process, the high-level radioactive waste will be essentially no different from what is being handled now—no new ground broken, no new qualifications in materials. The uranium also will be unchanged after it goes through the PUREX process. The same with plutonium; if it goes through the PUREX, you haven't changed the existing process. So people should not get excited about this new stuff coming in—we've handled it for fifty years.	2	Sections 2.3.2 and 2.5.4
The EIS should analyze blanket and driver fuels separately since they have different chemical and radiological characteristics and different treatments might be warranted for each.	6	Sections 2.5, 4.3, 4.4, 4.5, 4.6, 4.7, and 4.8.
We're glad to see the melt and dilute alternative, a nonseparation technology, is being considered in this EIS.	1	Sections 2.5.5, 4.6, 4.7, and 4.8
The EIS should not assume that everything is known about the C-22 canister's performance in all conditions that could affect disposal; therefore, this canister should not be the only type of containment considered for encapsulation.	1	Section 4.12
The EIS should clarify whether, if the PUREX process were used, the waste would remain at the Savannah River Site after processing or be returned to Idaho.	4	Section 4.5.6
The EIS must clarify whether DOE considers low-enriched uranium to be a waste.	1	Section 4.3
The EIS must clarify which specific waste form will be used before any spent nuclear fuel is treated.	2	Sections 4.2.6, 4.3.6, 4.4.6, 4.5.6, 4.7.6, and 4.8.6
Will all of the technology alternatives shown on the poster handout be evaluated in this EIS? Has DOE made the ultimate decision concerning which alternatives will be evaluated in this EIS?	1	Section 2.5
Is there anything different about handling the materials involved in this EIS that would make the chloride volatility alternative more viable than was found for aluminum enriched uranium fuels? Hasn't this alternative already been evaluated in another EIS?	1	Section 2.7
The chemistry of the electrometallurgical process and the other alternatives should be provided.	1	Appendix C

Issues	*No. of Comments*	*EIS References*
Blanket fuel can be mechanically declad and stripped of elemental sodium without the need for dissolution and separation of the solid fuel. While the minimal discussion in DOE documents stresses the difficulties of this approach, it is extremely hard to believe that the difficulties, costs, and risks of such minimal processing would be greater than those incurred by electrometallurgical treatment of the fuel. It is difficult to understand DOE's argument that this option is not as mature as electrometallurgical treatment, since it was employed for 15 times as many blanket rods as those that ultimately will be processed during the electrometallurgical treatment demonstration.	1	Section 2.5.3
Both DOE and the U.S. Nuclear Regulatory Commission (NRC) underplay the significance of the mechanical decladding of 17 metric tons of heavy metal of blanket fuel. NRC refers to this as a small amount even though it is 75 percent of the existing Experimental Breeder Reactor-II (EBR-II) blanket inventory. This is only one example of the loaded language in the Notice of Intent and its reference documents that strongly suggests the mechanical decladding alternative is not being fairly evaluated.	1	Section 2.5.3
All alternatives investigated and considered in this EIS should be viable and demonstrable. Unproven technologies preclude realistic bounding of environmental impacts and consequently do not appear to meet the intent of NEPA by providing implementable alternatives.	1	Section 2.5
Coordinate development of this EIS with others that are currently in preparation, including the Idaho National Engineering and Environmental Laboratory (INEEL) High Level Waste and Facilities Disposition, the Savannah River Spent Fuel, and the Yucca Mountain EISs.	3	Section 1.6
What are the plans for treatment of sodium-based fuel located at the other sites (about 2 percent of inventory)?	1	Section 2.2
Political decisions, such as the Idaho Settlement Agreement (which says that spent nuclear fuel must be out of Idaho by 2035), should not preclude any of the No Action Alternatives from being considered.	1	Sections 2.5.1, 4.2, and 4.12
I was pleased to hear you say you were looking at several options connected to the No Action [alternative].	1	Sections 2.5.1 and 4.2
The EIS should be specific about the stable compound of sodium and how that makes it like table salt (i.e., not a problem).	1	Appendix C and Section 2.3
How does this EIS relate to other EISs for treatment and disposal of other spent nuclear fuel types?	1	Section 1.6
What is the enrichment of the uranium?	1	Section 2.2.1
DOE should consider whether adequate information exists to allow estimation of bounding impacts for at least one treatment alternative in addition to the PUREX process at the Savannah River Site, the proposed electrometallurgical treatment at Argonne National Laboratory-West (ANL-W), and the No Action Alternative. Instead of dismissing various treatment alternatives from further analysis, DOE should use existing information about those alternatives to support evaluation of as many treatment alternatives as possible. For example, the processing experience at Idaho Nuclear Technology and Engineering Center (INTEC) of the driver fuel using the PUREX-type process might be used in the analysis of the PUREX process at Savannah River.	1	Sections 2.5.3, 2.5.5, 4.4, 4.6, 4.7, and 4.8
To support public review of the alternatives under consideration, the EIS should offer complete descriptions of how each alternative would be implemented.	1	Appendix C and Section 2.3

Issues	*No. of Comments*	*EIS References*
Each alternative should include full descriptions of all materials (including wastes) resulting from treatment; proposed handling of all materials used in the treatment process; environmental impacts; measures to provide environmental protection; measures to ensure worker and public safety; facilities needed; full and complete discussion of waste handling facilities, magnitude and characteristics of the waste streams, type and amount of storage, and ultimate disposal method and location.	1	Sections 4.2, 4.3, 4.4, 4.5, 4.6, 4.7, and 4.8
The EIS should provide bounding estimates of the size, frequency, and number of expected shipments of products leaving Idaho on an annual basis.	1	Section 4.10
The EIS should provide bounding estimates of the duration of time that INEEL would store any products before shipment elsewhere after treatment.	1	Sections 4.2.6, 4.3.6, 4.6, 5.6, 7.1, and 8.0
Preparation of the EIS and the related decision-making process should be coordinated with related environmental documentation being prepared to ensure they are based on common data and common planning assumptions.	1	Section 1.6
The EIS should deal with disposition of all the waste streams resulting from this proposed action.	2	Sections 2.8, 4.2.6, 4.3.6, 4.4.6, 4.5.6, 4.6.6, 4.7.6, and 4.8.6
To help the public understand DOE's rationale for moving forward with this decision, the EIS should describe how each treatment alternative would address the waste acceptance criteria for resulting waste products destined for disposal at current and planned disposal facilities.	1	Sections 2.8 and 4.12
The Draft EIS should include a complete subject index and not just an alphabetically arranged list of headings.	1	Chapter 9
DOE should coordinate the related projects [e.g., the Idaho High-Level and Facilities EIS; the Management of Savannah River Spent Nuclear Fuel EIS; and the Geological Disposal Repository for Spent Nuclear Fuel and High-Level Waste at Yucca Mountain, Nevada, EIS] to support consistent, coordinated decision-making.	1	Section 1.6

Additional issues were added to the scope of the EIS as a result of the public scoping process. These issues are listed in **Table A-2**.

Table A–2 Issues Added to the Scope of the EIS

Issues	*No. of Comments*	*EIS References*
Analyses related to the No Action Alternative should include the environmental consequences of not doing anything...and [this alternative] should not be written off because somebody made a political decision that this stuff will be out of Idaho by 2035.	1	Section 4.2
The proposed structure of the EIS as described in the Notice of Intent is inconsistent with DOE's approach to spent nuclear fuel management at other sites and prematurely promotes a preferred option for managing sodium-bonded spent nuclear fuel. By presuming the proposed action is electrometallurgical treatment, the proposed structure of the EIS effectively establishes this treatment as the preferred alternative for stabilization of this material. While it is reasonable to rule out obviously impractical alternatives in the scoping process, several of the alternatives described in the Notice of Intent are technically viable and should not be prematurely dismissed.	3	Sections 1.2, 1.3, 1.4, and 2.5

Issues	*No. of Comments*	*EIS References*
DOE should consider the possibility of using different treatment processes for treatment of the driver fuel and the blanket fuel. Could the driver fuel be handled as part of the ongoing demonstration? Treatment alternatives for the blanket fuel could conceivably include direct disposal, as it is not yet clear that it will require treatment before disposal.	1	Sections 2.5.3, 2.5.4, 2.5.5, and 2.5.6
The three alternatives presented for treatment of the EBR-II fuel are the most reasonable ones politically available, namely (1) separate the highly enriched uranium and make the other materials into a ceramic using a hot isostatic press, or (2) separate both the uranium and plutonium using the PUREX process at the Savannah River Site and...vitrify the wastes, or (3) direct burial.	1	Sections 2.5, 4.2, 4.3, and 4.4

DOE responded to all issues raised during the scoping period. Many of the public issues were not analyzed for a specific reason or were determined to be outside the scope of the EIS. These issues are listed in **Table A–3**. Corresponding responses from DOE also are provided in Table A–3 to explain why each issue was not analyzed.

Table A–3 Other Issues Considered

Issues	*No. of Comments*	*DOE Responses*
Costs		
The public needs information about the cost of the proposed action and the costs of the other technology alternatives before it can adequately comment on the EIS.	6	Information on cost will be made available to the public via the Cost Analysis Report, which will be issued during the Draft EIS public comment period.
This program is not worth the money it will cost.	1	Information on cost can be found in the Cost Analysis Report which, along with the EIS, will factor into the Record of Decision.
The cost assessment has to be part of the EIS.	2	Although the cost assessment is not part of the EIS, it has been prepared concurrently with the EIS. The Cost Analysis Report, along with the EIS, will factor into the Record of Decision.
If you don't account for the low-enriched uranium stream, your cost estimates are going to be wrong or at least off. If you don't have a disposition scenario, you have to look at the long-term economic and environmental storage costs that will belong to DOE for a long time.	2	The environmental impacts and cost of storage of the low-enriched uranium stream have been analyzed in the EIS and Cost Analysis Report, respectively.
We think that combining the research and development efforts on these two different types of fuel [blanket and driver] might lead to considerable cost savings.	1	If an alternative technology is chosen that could treat both the driver and blanket fuel, research and development efforts would be combined, as they were for electrometallurgical treatment research and development.
As Savannah River has a huge vitrification facility and that technology already is available, DOE should compare the costs of vitrification with the costs of the PUREX process.	1	The vitrification facility at the Savannah River Site treats the high-level radioactive waste that results from PUREX processing. The two are not independent. The cost of vitrification will be included in the cost of the PUREX alternative in the Cost Analysis Report. Direct vitrification of sodium-bonded spent nuclear fuel, however, is not technically feasible.

Issues	*No. of Comments*	*DOE Responses*
Cost analysis should include: (1) program costs so far in detail, including whether these costs were for pyroprocessing or for the EBR-II to shut down; (2) how much it would cost to close out the program at the end of the test, including decommissioning the machinery and dealing with all the waste streams (such as low enriched uranium); (3) what it would cost to scale-up the program, including commissioning and dealing with all waste streams at the end of the scale-up.	1	The Cost Analysis Report does not include EBR-II shutdown costs. The Cost Analysis Report includes the cost of any new machinery, if needed; treating the sodium-bonded spent nuclear fuel; deactivating machinery; and dealing with the waste streams. The low-enriched uranium product is not a waste. Its disposition will be the subject of a future NEPA review, however, the cost of storage of the low-enriched uranium is included in the cost analysis report.
The EIS should include the cost of transportation if this stuff is moved across country from Idaho to South Carolina and then from South Carolina to wherever.	1	The cost of offsite and onsite transportation is included in the Cost Analysis Report.
Environment, Safety, and Health		
The Shoshone-Bannock Tribe considers the INEEL land to be part of their original territory and believes the electrometallurgical treatment process will not impact the land's cultural resources or native species and will make the best uses of these resources.	1	The commentor's support for the electrometallurgical technology is acknowledged.
DOE should explain the environmental considerations that are pushing this EIS to completion in such a short period of time, including the environmental threats of continuing to store the EBR-II spent nuclear fuel in Idaho, if any. Then, DOE should compare these environmental threats with the R&D schedule for all the alternative technologies being considered, especially the nonseparation technologies.	1	The purpose and need for agency action is discussed in Section 1.2. Under the No Action Alternative, the Department may decide to continue to store the sodium-bonded spent nuclear fuel indefinitely, or until research and development of an alternative treatment technology is successfully completed.
DOE should be able to provide the environmental impacts for all of the alternative technologies listed in the Notice of Intent; they should not be dismissed because DOE does not know enough about them.	1	Alternative technologies were not dismissed solely based on the lack of available information on the respective technologies. As discussed on Section 2.6, chloride volatility was dismissed due to the potentially significant (in comparison to other treatment technologies) occupational and public risks from the volatilization of fission products and chloride gas.
Nonproliferation		
Nonproliferation should not be addressed in a separate report; the nonproliferation assessment should be part of the EIS. Short-circuiting the nonproliferation analysis is particularly egregious in light of the pledge in the Notice of Intent to include this assessment in the draft EIS and the existence of such a DOE assessment from December 1998.	3	The Notice of Intent stated, " The combination of the information contained in the Draft EIS, the public comment in response to the Draft EIS, and the nonproliferation impacts assessment report will enable the Department to make a sound decision..." Although the nonproliferation report is separate from the EIS, it will fully analyze the nonproliferation impacts of the alternatives in the EIS.
The public should have an opportunity to comment on the ongoing nonproliferation assessment, and the assessment should be publicly available before the comment period is closed on this EIS.	9	The report will be available to the public prior to the end of the public comment period for this Draft EIS. However, the nonproliferation report will be issued as a final document.

Issues	*No. of Comments*	*DOE Responses*
The public needs information about the nonproliferation impacts of the proposed action before it can comment on the EIS.	1	The nonproliferation report will be available to the public prior to the end of the comment period for this Draft EIS.
The EIS should not be released until nonproliferation concerns no longer are being debated; there is a potential for exporting this technology.	1	The nonproliferation report will be available to the public prior to the end of the comment period for this Draft EIS.
Given that obtaining fuel material is the greatest hurdle to producing nuclear weapons, DOE should take nonproliferation concerns about small-scale reprocessing technologies like pyroprocessing more seriously and give them greater weight in its decision-making.	2	DOE is concerned with the nonproliferation impacts of all of its proposed actions. It is for this reason that a separate nonproliferation impacts assessment report will be prepared specifically to address the alternatives under consideration.
Pyroprocessing is a proliferation-prone technology. For example, although plutonium no longer would be separated as a separate step in the EBR-II treatment, the original pyroprocessing technology was intended to remove plutonium and actinide components in a liquid cadmium cathode, and that option is always there.	4	DOE has conducted four independent nonproliferation assessments of electrometallurgical technology over the past 11 years. A new assessment that addresses the alternatives under consideration for treating sodium-bonded spent nuclear fuel is being conducted concurrently with the EIS and the report will be available for public review. Previous assessments have concluded that electrometallurgical technology was not capable of separating plutonium in a form that would be suitable for weapons. Development of the liquid cadmium cathode was canceled before significant engineering issues were resolved. No liquid-cadmium cathode was ever completed for the electrorefiners used in the Fuel Conditioning Facility, where the spent nuclear fuel treatment would take place under the preferred alternative.
Pyroprocessing will continue to search for other missions before the issue of whether it can be shut down and decommissioned on a timely basis is decided. Use of pyroprocessing should be “nipped in the bud” because of nonproliferation concerns.	1	Electrometallurgical treatment technology is a promising technology for the management of spent nuclear fuel. DOE is considering applying this technology for the management of some or all of its sodium-bonded spent nuclear fuel at sometime in the near future. DOE is conducting a nonproliferation assessment that focuses on the application of electrometallurgical and alternative treatment technologies to sodium-bonded spent nuclear fuel. This new assessment will be made available to the public during the Draft EIS public comment period. Previous nonproliferation assessments have found electrometallurgical technology to be in accordance with the U.S. nuclear nonproliferation policy for the specific applications considered.
The Savannah River nonproliferation assessment states that pyroprocessing can be modified to produce plutonium. This modification may not be easy, but it would be easier than building an entire PUREX facility or adding such a capability to any of the other nonseparation technology options—and it would certainly be of interest to rogue states who are interested in producing nuclear weapons.	3	The modification referred to in the Savannah River nonproliferation assessment involves adding a proven aqueous process such as PUREX onto the electrometallurgical process. Because the aqueous processes would be incompatible with the dry inert atmosphere required by the electrometallurgical process, a separate facility would be required. If a nation bent on weapons production had this capability, it could separate weapons-usable plutonium directly from spent nuclear fuel or plutonium production targets without the need for the electrometallurgical process equipment.
This program is inconsistent with the present U.S. position on reprocessing. The United States should not be funding new separation technologies.	2	The DOE Office of Arms Control and Nonproliferation will assess the nonproliferation impacts of the alternative treatment technologies under consideration in this EIS in a separate report to determine if the alternatives are consistent with U.S. nonproliferation policy and goals.

Issues	No. of Comments	DOE Responses
Pyroprocessing is reprocessing. MacArthur Prize Fellowship winner Frank Von Hippel and Professor James Warf, inventor of several reprocessing technologies, underscore this fact and express concern about the nuclear nonproliferation impacts of pyroprocessing: "...because pyroprocessing facilities are more compact than conventional facilities, they are easier to conceal. The world would become a more dangerous place."	2	In a nonproliferation assessment conducted for DOE in 1992, a panel of experts stated that there was no reason to conclude that electrometallurgical process facilities would be any easier to conceal than a conventional reprocessing plant. The electrometallurgical process requires a large heavily shielded hot cell with highly purified argon atmosphere and specialized process equipment.
While the Notice of Intent states that DOE has no plans to apply this technology (electrometallurgical treatment) to any other types of spent nuclear fuel, it clearly leaves the door open for other applications and raises the concern that ANL-W will continue to hunt for other materials that can be used to keep the electrometallurgical treatment apparatus operating after the sodium-bonded fuel campaigns are completed, or even to justify construction of new facilities. This open-ended approach... has severe implications for nonproliferation.	1	Electrometallurgical treatment technology is a promising technology for the management of spent nuclear fuel. DOE is considering applying this technology for the management of some or all of its sodium-bonded spent nuclear fuel at sometime in the near future. DOE is conducting a nonproliferation assessment that addresses the application of electrometallurgical technology, as well as the other alternatives under consideration, to sodium-bonded spent nuclear fuel. This new assessment will be made available to the public during the Draft EIS comment period. Previous nonproliferation assessments have found electrometallurgical technology to be in accordance with U.S. nuclear nonproliferation policy for the specific applications considered.
The electrometallurgical treatment process can be modified to produce plutonium. Moreover, there are no plans to place ANL-W facilities under international safeguards. Therefore, from an arms control standpoint, the Fuel Conditioning Facility must be regarded as a dual-use facility capable of being operated as a reprocessing plant. In view of this, it is highly advisable to prepare for timely shutdown of the facility when any campaigns for which it is determined to be essential (if any) are completed.	1	DOE has conducted four independent nonproliferation assessments of electrometallurgical technology. A new assessment that focuses on the application of electrometallurgical technology to sodium-bonded spent nuclear fuel is being conducted concurrently with the EIS and will be available for public review. Previous assessments have concluded that electrometallurgical technology was not capable of separation plutonium in a form that would be suitable for weapons. Development of the liquid cadmium cathode was canceled before significant engineering issues were resolved. No liquid-cadmium cathode was ever completed for the electrorefiners used in the Fuel Conditioning Facility, where the spent nuclear fuel treatment would take place. The Fuel Conditioning Facility operates under DOE safeguards and security requirements.
DOE should make the nonproliferation assessment of the proposed electrometallurgical treatment action a part of the NEPA process. The assessment should cover not only the proposed action, but the broader proliferation implications of continued research and development of this reprocessing technology.	1	DOE is concerned with the nonproliferation impacts of all of its proposed actions. It is for this reason that a separate nonproliferation impacts assessment report will be prepared that will specifically address electrometallurgical treatment technology. DOE will consider this report in its decision-making process.
One issue that should be covered in the nonproliferation assessment is whether promotion of electrometallurgical treatment as a "proliferation-resistant" technology ultimately will prove harmful to U.S. nonproliferation goals. If this designation does not have a sound technical basis (as we believe it does not), the ultimate result will be an increased danger of proliferation.	1	DOE is concerned with the nonproliferation impacts of all of its proposed actions. It is for this reason that a separate nonproliferation impacts assessment report will be prepared that will specifically address electrometallurgical treatment technology.

Issues	*No. of Comments*	*DOE Responses*
For nations that reprocess spent nuclear fuel, switching to electrometallurgical treatment may enable them to argue that their current safeguards burden should be relaxed.	1	Prior to the export, to a foreign nation, of any technology that may have nonproliferation impacts, the Department assesses the impacts, if any, to ensure that U.S. nonproliferation goals are met.
The EIS should include a detailed, thorough analysis of the weapons proliferation implications of each treatment alternative.	1	DOE's Office of Arms Control and Nonproliferation is preparing a report on the proliferation implications of each treatment alternative. This new assessment will be made available to the public during the Draft EIS public comment period.
One of the justifications for proceeding with the mixed oxide (MOX) proposal was to satisfy the international community's desire to forestall the ready availability of weapons-grade materials. This proposal creates the ready availability of those same materials. The EIS must account for this apparent contradiction of policy and address the measures intended to safeguard the byproduct(s) of this process.	1	The Department recognizes the need to identify nonproliferation impacts of the treatment technologies. Therefore, the DOE Office of Arms Control and Nonproliferation will assess the nonproliferation impacts of the alternative treatment technologies in a report, separate from this EIS.
Alternative Technologies		
The EIS should re-evaluate and address plutonium separation; it would be less expensive to separate the plutonium because that would mean the repository would need to last only 300 years, instead of 10,000.	1	The EIS is evaluating plutonium separation as a part of the PUREX option for the blanket fuel. Plutonium separation would not guarantee a different performance requirement for the repository, since the long-term requirements are driven by other radioisotopes.
DOE has already made up its mind. Other methods than pyroprocessing haven't been given sufficient attention. These alternative methods continually are slated as "not developed enough." Yet in three years, there hasn't been much attention given to developing them to a point where they could be reviewed fairly. Alternative new technologies should not be dismissed due to lack of knowledge about them.	4	In response to public comments, DOE has reformulated the scope of the EIS to address more generally the treatment and management of DOE sodium-bonded spent nuclear fuel. Information developed in the course of preparing this EIS suggests that alternative technologies may have certain advantages (e.g., cost) for some or all of the fuel. Accordingly, DOE has no preferred alternative at this time. With respect to less developed technologies, in the EIS DOE is considering an option under the No Action Alternative in which the Department would actively conduct research and development of promising new technologies.
The Notice of Intent is biased toward electrometallurgical treatment because it disparages the other alternatives, which are tacked on just to satisfy a legal requirement. The program is taking the wrong approach toward electrometallurgical treatment because the alternatives are not really valid.	2	In response to public comments, DOE has reformulated the scope of the EIS to address more generally the treatment and management of DOE sodium-bonded spent nuclear fuel. Information developed in the course of preparing this EIS suggests that alternative technologies may have certain advantages (e.g., cost) for some or all of the fuel. Accordingly, DOE has no preferred alternative at this time. With respect to less developed technologies, in the EIS DOE is considering an option under the No Action Alternative in which the Department would actively conduct research and development of promising new technologies.
There is a danger that other technologies will be abandoned if, as it appears, DOE is rushing to produce waste or materials to go to a waste site somewhere or is pushing pyroprocessing ahead of other technologies.	1	In response to public comment, DOE has restructured the alternatives to be considered, including an option of deferring a treatment decision and developing alternative technologies.
The EIS should identify the alternative sites if Idaho is not selected and which sites will be needed for the alternative technologies.	1	The EIS has identified the Savannah River Site as an alternative site for the PUREX and melt and dilute alternatives.

Issues	*No. of Comments*	*DOE Responses*
The EIS should include a stabilization timeline on environmental grounds for EBR-II spent nuclear fuel. The timeline should include the time needed to more fully develop other alternatives.	2	EBR-II spent nuclear fuel must be removed from the State of Idaho by the year 2035 in accordance with a DOE/State of Idaho Settlement Agreement, signed in October of 1995. DOE believes that treatment to remove sodium from EBR-II and other spent nuclear fuel will make acceptance of this fuel in a national geologic repository much more likely.
Will the EIS look at the vitrification facility at INTEC?	1	The proposed Vitrification Facility at INTEC is not compatible with any of the proposed waste forms or metal fuel such as the EBR-II or Fermi-1 fuel. It is for this reason that DOE has not analyzed this facility in the EIS.
The EIS should address the size of the electrometallurgical treatment facility and whether the plant capacity is greater than needed for the proposed mission (more than 62 metric tons of heavy metal).	1	The plant capacity of the electrometallurgical treatment equipment as described in the preferred alternative is approximately 5 metric tons of heavy metal per year. It would therefore require 12 years to treat the entire 60-metric ton DOE sodium-bonded spent nuclear fuel inventory.
The Notice of Intent indicates that DOE has no plans to apply electrometallurgical treatment to any other spent nuclear fuel types, suggesting the plant would be decommissioned after completing the electrometallurgical treatment mission for sodium-bonded spent nuclear fuel. The EIS, therefore, should address the impacts of decommissioning the plant.	2	At this time, DOE has no intent to apply electrometallurgical treatment to any other spent nuclear fuel types. The electrometallurgical treatment process equipment is housed within a large multipurpose hot cell facility which has programmatic value to DOE, even in the absence of a spent nuclear fuel treatment program. Any specific electrometallurgical treatment equipment would be deactivated at the end of any treatment program; however, there are no plans to discontinue use of the hot cell facility.
Use a reactor or accelerator to fission the transuranic material.	1	This is not a reasonable alternative because the transuranic materials resulting from the electrometallurgical treatment process would require extensive additional processing before they would be suitable for fission in a reactor.
Adding another furnace and cathode to ANL-W's facility would both accelerate the processing and provide opportunities for new research.	1	The existing electrometallurgical treatment equipment would provide DOE an adequate processing rate for the sodium-bonded spent nuclear fuel inventory. New research would be accomplished with equipment in a nonradioactive laboratory environment.
Regarding the use of melt and dilute and Savannah River—the Savannah River process will not be sized or configured to handle INEEL fuels (which should be contrary to the Foreign Research Reactor Record of Decision). Melt and dilute at INEEL solely should be the alternative.	1	The sodium-bonded fuel would have its cladding and sodium removed before being placed in aluminum cans for shipment to the Savannah River Site, where the proposed melt and dilute process would take place. This pretreatment step would make the fuel compatible with the proposed Savannah River Site process.
Sodium is highly reactive with water/moisture, and this property could be taken advantage of by controlled reaction on a limited scale—exposing the sodium-bonded material to moisture. The sodium hydroxide formed could be neutralized with an appropriate acid, allowing the remaining spent nuclear fuel to loose its pyrophoric properties. Please address this in the EIS.	1	For those fuels in which the sodium can be exposed, the EIS describes a process for safely removing it by vacuum distillation. The process described in the comment would accelerate corrosion of the uranium, resulting in an unsafe pyrophoric condition.
DOE may want to consider an alternative that examines the relationship between the EBR-II fuel at INEEL and the high-level radioactive waste at the stabilization facility.	1	The proposed INEEL high-level radioactive waste management EIS is considering methods to manage the calcine that was produced from the reprocessing of DOE spent nuclear fuel at INTEC. With the decision to shutdown the reprocessing facilities, no processes are currently available that would make the sodium bonded fuel compatible with the calcine.

Issues	*No. of Comments*	*DOE Responses*
The fall 1996 National Research Council report on pyroprocessing at ANL states that even more time and money than originally planned will be needed to "achieve the program's objectives" and raises troubling questions about several aspects of the research itself. Later reports, unfortunately, do not specifically follow up on these concerns.	1	The Demonstration Project has addressed concerns that have been raised by the National Research Council. Their 1998 report has recognized the progress in the Demonstration and has stated it should continue to completion.
The fall 1996 National Research Council report raises serious concerns about several aspects of the research including a lack of coordination between ANL East and West. This lack of coordination and differing goals have led to duplicate efforts in at least one case and equipment failures. The report notes the lack of a "well-coordinated implementation plan between ANL East and West...."	1	The electrometallurgical demonstration project, which is nearing completion at ANL-W, has successfully met National Research Council criteria to date. The success of this demonstration project has been possible only through close coordination between scientists and engineers at ANL East and West.
The [fall 1996 National Research Council] report found that equipment is not performing at expected levels and separation efficiencies are lower than expected. This means that, so far, the basic goal of the pyroprocessing program—to separate the uranium from the rest of the irradiated fuel—has not been met.	1	In the spring 1998 status report, the National Research Council recognized the progress made in the demonstration and recommended that the demonstration be carried to completion.
Research on selected alternatives should have been carried out to support a defensible analysis of their feasibility in the EIS.	1	The alternatives to be analyzed in detail are described in Chapter 2 of the EIS. An analysis of their feasibility is included in this chapter.
DOE has not demonstrated there is a safety-based need to process the driver fuel by experimentally assessing the impact of elemental sodium on radionuclide leach rates.	1	DOE has proposed treatment to remove the sodium from sodium-bonded spent nuclear fuel to allow acceptance of this fuel in a national geologic repository. This is because sodium reacts with water in the environment to form corrosive sodium hydroxide solutions and potentially explosive hydrogen gas.
DOE should initiate a process similar to the Processing Needs Assessment to determine at the earliest possible date the "small quantities of certain spent nuclear fuels" that may be considered for electrometallurgical treatment in the future. Such an effort is essential for shutdown and decommissioning planning.	1	At this time DOE has no intent to apply electrometallurgical treatment to any other spent nuclear fuel types. If, during the sodium-bonded fuel treatment program, DOE finds another application for electrometallurgical treatment at ANL-W, the development of plans to deactivate the electrometallurgical treatment equipment at ANL-W would be delayed accordingly.
A study similar to the 1997-98 Processing Needs Assessment should be conducted to identify all materials in the DOE complex that might need reprocessing in the Savannah River Site canyons for stabilization purposes, thus limiting the universe of potential uses for the canyons and facilitating planning for their shutdown. A similar process should be conducted for the Fuel Conditioning Facility as part of this EIS process, with the opportunity for full public participation and comment.	1	The EIS is being coordinated with other DOE EIS documents and Records of Decision concerning complex-wide management of spent nuclear fuel. These EISs are described in Section 1.6 of this EIS.

Issues	*No. of Comments*	*DOE Responses*
It is unfortunate that the option of separating the plutonium along with the uranium by the electrometallurgical process could not have been considered. Although the resulting fissile material would only have been suitable for a fast-neutron reactor...at least we would not have the agony of worrying about putting this plutonium in a repository.	1	The electrometallurgical process cannot separate plutonium. Because of potential nonproliferation implications, the Department elected not develop the capability for electrometallurgical processing to produce any plutonium-bearing product. Plutonium separation is an integral part of alternative 3, PUREX processing of the blanket fuel at the Savannah River Site. However, removal of the plutonium would not significantly affect the long-term performance of the repository, which is driven by other radioisotopes.
Since the electrometallurgical method works, is ready to go, and is not expensive, it is in the public interest to get the fuel treatment job done rather than delay while developing some other method.	1	The commentor's support of the electrometallurgical treatment technology is acknowledged.
The addition of depleted uranium to the electrometallurgical treatment process is both a waste of depleted uranium and enriched uranium. Why add the depleted uranium?	1	Blending depleted uranium with the highly enriched uranium recovered from the spent EBR-II driver fuel results in low-enriched uranium. This step, which is consistent with U.S. nonproliferation policy, results in lower costs for storing and safeguarding the uranium. Because the uranium ingots still contain more enrichment than is required for commercial power reactor fuel, their potential economic value is not decreased. The Department currently stores more than 500,000 tons of depleted uranium for which no immediate use is planned. Using some 10 tons of this inventory for treating spent nuclear fuel would have no discernable impact.
Waste		
The EIS should address the disposal specifications for spent nuclear fuel, and DOE should make sure that, whatever technology is selected, the spent nuclear fuel will meet repository specifications. This determination should be made before the canyons are shut down to avoid precluding a way to get rid of the materials.	1	The ceramic and metal high-level radioactive waste forms that would be produced from the proposed action are expected to be at least as durable as the borosilicate glass high-level radioactive waste form. The design criteria for the national spent nuclear fuel repository include receipt and disposal of the borosilicate glass high-level radioactive waste.
The EIS should explain why stainless steel and noble metals are considered wastes and not potentially valuable resources.	1	The stainless steel and noble metals would be part of the metal high-level radioactive waste forms. High-level radioactive waste is a material that NRC has determined requires permanent isolation.
Waste characterization is a problem. Low enriched uranium is a problem-it's a waste not a product. The EIS should look at the long-term storage costs of uranium.	2	DOE does not consider low-enriched uranium to be a waste. No highly enriched uranium would result from any of the alternatives considered at INEEL.
Discussion of the low-enriched uranium stream must include a full analysis of what happens to this stream and when.	1	DOE has not made a decision concerning future uses for the low-enriched uranium other than that the low-enriched uranium would not be used for defense purposes.
Spent nuclear fuel is not a waste.	1	Spent nuclear fuel is a fuel that has been withdrawn from a nuclear reactor following irradiation; the constituent elements have not been separated for reprocessing.

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The project is being sold as a way to reduce the volume of waste to Yucca Mountain. It won't reduce actual volume; it will only increase floor space by putting ceramic and metal waste forms closer together while still avoiding criticality issues. That's where your 65 percent comes from. You don't have volume reduction; you just have split the waste into lots of different forms which you still have to find a home for. But the message that is getting out is that you will be sending a smaller by weight number of packages to Nevada.	3	Waste volumes, masses and disposal paths for all types of wastes are considered for the different alternatives in this EIS. The volume of high-level radioactive waste or spent nuclear fuel that would be sent to a geologic repository are some of the things considered in the waste management sections. The potential impact on different disposal sites is considered and discussed. However, the purpose and need for the proposed action is to treat and manage the spent fuel, not to reduce the volume of waste that eventually will be sent to a repository.
DOE does not know if electrometallurgical treatment wastes will meet the repository waste acceptance criteria. DOE does not know what those criteria will be—or if there will be any repository at all. Will the waste be acceptable? We need honest assumptions on the waste stream.	4	The repository waste acceptance criteria are still being developed. However, the ceramic and metal waste forms that would result from the electrometallurgical treatment process are expected to be accepted into the repository.
DOE should consider dealing with this high-level radioactive waste as part of the high-level radioactive waste being dealt with at INTEC.	1	The proposed INEEL high-level radioactive waste management EIS is considering methods to manage the calcine that was produced from the reprocessing DOE spent nuclear fuel at INTEC. With the decision to shutdown the reprocessing facilities, no processes are currently available that would make the sodium bonded fuel compatible with calcine. The restart of these facilities was considered and eliminated from the alternatives.
DOE admits to having no knowledge of the whereabouts of the documents pertaining to previous removal of the sodium bonding from 17 metric tons of EBR-II blanket fuel via mechanical decladding. Such mismanagement, if true, is of concern and should be investigated. We request that a greater effort be undertaken to find these documents and make them publicly available during the EIS period.	1	DOE has found the documents that describe the process, equipment, operating procedures and waste disposal paths for the decladding and sodium removal of the 17 metric tons of EBR-II blankets. These documents were considered during the selection of the proposed decladding and sodium removal alternatives.
DOE's plans for disposing of the low-enriched uranium created from this process—will it be stored as a waste or sold as a resource?	2	DOE has not made a decision concerning future uses for the low-enriched uranium produced by the electrometallurgical treatment other than the decision that the low-enriched uranium would not be used for defense purposes.
This program [electrometallurgical treatment] has no place in a sound nuclear waste management policy. Proponents of this program are.....making the problem worse not better. This program will increase the complexity and amount of nuclear waste generated at ANL. We do not support an expansion of this program and urge that it be terminated.	1	DOE believes that treating sodium-bonded spent nuclear fuel is in keeping with sound nuclear waste management. This is because the proposed action would reduce uncertainty regarding waste disposal. Also, the number of canisters that must be disposed of in a geologic repository is reduced. Further, ceramic and metal waste material is very durable and has been formulated to be unreactive in the environment.
If DOE creates high-level radioactive waste in a vitrified form, there will be three forms of high-level radioactive waste in one Idaho county (ceramic, metal, vitrified).	2	The statement is correct. Different waste streams often require different stabilization techniques. The ceramic, metal and vitrified waste forms are being developed because they are best suited for specific waste streams.

Issues	*No. of Comments*	*DOE Responses*
If this material won't meet the disposal specifications for the repository, a specification should be incorporated into the Record of Decision to say that DOE will look at this material and its proposed specifications before the canyons are shut down to ensure it is as good as the PUREX borosilicated glass that is being prepared for the Yucca Mountain repository.	1	DOE will consider the programmatic impacts including schedule and technical uncertainties such as availability and waste acceptance when a Record of Decision is made.
Since the waste acceptance criteria at Yucca Mountain currently is not confirmed, how do you intend to meet and store [the waste] for "road-ready" conditions?	1	The present goal is to place the spent nuclear fuel and high-level radioactive wastes at ANL-W in retrievable storage so that it can be shipped to the proposed packaging facility that will ship the INEEL DOE spent nuclear fuels to the repository. For the Savannah River Site alternatives, the high-level radioactive waste glass or melt and dilute product will be coordinated with these streams that will be produced at Savannah River Site.
Will planned dry storage have to be retreated later to meet acceptance criteria at Yucca Mountain?	1	The No Action alternative may require future treatment. The goal of the other alternatives is to put the waste in road ready condition without further treatment. The uncertainty in the final repository waste acceptance criteria is part of the programmatic considerations.
Uranium metal also is reactive; will it be treated before placement in a geologic depository?	1	Uranium metal is currently managed as part of the materials disposition program and is out of the scope of the EIS.
The Environmental Assessment contained ridiculous estimates of waste streams, especially the low-level radioactive waste streams. Actual information about wastes generated from the demonstration project should be released to the public for use in the EIS.	1	The actual waste generation rates for the demonstration project have been used to calculate estimates of waste streams in this EIS.
Previous National Research Council reports have concluded that several of the waste forms generated by this technology [pyroprocessing] would not be suitable for placement in a geologic repository. The fall 1996 National Research Council report raises serious concerns about the testing procedures used to determine whether one of the new waste forms will be suitable for placement in a geologic repository. Most troubling of all is the analysis of ANL's choice of test protocol. A key issue is the release of the radionuclides from the waste. The report notes that the test protocol focuses on a radionuclide release mechanism that is... "incorrect at best, and potentially misleading at worst."	1	In order to address the question on waste form qualification, DOE has asked the National Research Council to conduct a specific review on this subject. The report that discusses the results of this waste qualification review and the other NRC reports will be considered when a record of decision is formulated.
Since getting waste ready for a geologic repository is the justification for this project, it must not go forward until the waste produced by the demonstration project has been fully characterized, which will occur early in the next century.	1	The uncertainty and status of each waste or spent nuclear fuel characterization are part of the programmatic consideration when a record of decision is formulated.

Issues	*No. of Comments*	*DOE Responses*
Spent nuclear fuel must be removed by 2035 as a result of processing. One concern is that transuranic waste will go to the repository, but low-enriched uranium and highly enriched uranium will stay at INEEL.	1	No highly enriched uranium would result from any of the alternatives considered at INEEL. DOE has not made a decision concerning future uses for the low-enriched uranium other than the decision that the low-enriched uranium would not be used for defense purposes. DOE will compare all reasonable alternatives on the basis of cost, including the cost of long-term storage of materials.
Compare heat loading with the ceramic and metal waste forms to heat loading of the highly enriched uranium rods—are they comparable with commercial spent nuclear fuel?	1	As packaged for disposal in a geological repository, the heat loading for the ceramic and metal waste forms are higher than that for the highly enriched uranium fuel because of fissile material limits for disposal packages. These high-level radioactive waste packages in general have lower heat loads than commercial spent nuclear fuel. Heat load would not be a concern regarding potential disposal in a geologic repository.
Transportation		
These materials should not be transported throughout the United States.	1	It is DOE's intention to minimize transport of radioactive materials associated with its sodium-bonded spent nuclear fuel inventory wherever possible.
If the ultimate burial place for the high-level radioactive waste is 1,000 miles away instead of 2,000 miles away, is that fact insignificant to transportation?	1	Generally, the environmental impacts of transporting spent nuclear fuel and high-level radioactive waste are small and would not differ significantly under the example posed by the commentor. DOE recommends the commentor see the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* for additional information on this subject.
The EIS should evaluate the potential for terrorism, especially during transportation.	2	The potential for terrorist acts involving material transports does not fall within the scope of this EIS.
Is it not known that, if the waste is sent to South Carolina [SRS], it will have to go somewhere else eventually; it won't stay in South Carolina?	1	As described in Section 2.5 of the EIS, Alternatives 3 and 5 would result in the storage of wastes or byproducts at SRS in South Carolina. For Alternative 3, the products from processing blanket fuel in the PUREX facility would be plutonium metal, borosilicate glass logs, and depleted uranium . For Alternative 5, the metal waste product from the blanket fuel melt and dilute process would be stored in the L Area at the Savannah River Site.
The EIS should provide bounding estimates of the size, frequency, and number of expected shipments of products coming into Idaho.	1	Chapter 4 and Appendix G of the EIS provide estimates of the size, frequency, and number of expected shipments of products coming into Idaho. The Record of Decision for the 1995 *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* also describes the size, frequency, and number of spent nuclear fuel shipments coming to Idaho.
DOE should develop an agreement with the Shoshone-Bannock Tribes to allow and appropriately manage the transport of any radioactive materials across the reservation.	1	Regardless of the alternative chosen, DOE will proceed in accordance with the DOE/Shoshone-Bannock Tribes Agreement-in-Principle, which covers notification and coordination of the transport of radioactive materials across the Fort Hall Reservation.

Issues	*No. of Comments*	*DOE Responses*
EIS Schedule		
This EIS may not be needed because the 1996 Environmental Assessment may be adequate.	1	DOE prepared an Environmental Assessment for the demonstration of electrometallurgical treatment on a limited amount(1.6 metric tons) of sodium-bonded spent nuclear fuel. In the May 15, 1996 Finding of No Significant Impact for the Environmental Assessment, DOE committed to prepare an EIS before applying the electrometallurgical treatment technology to the production-scale treatment of the sodium-bonded spent nuclear fuel inventory.
The Draft SBSNF EIS should not be issued for public comment before publication of relevant reports (e.g., waste qualification) from the National Research Council or the ongoing nonproliferation study. The schedule implies that DOE is not interested in incorporating the results from these studies into the EIS. Therefore, the timeline for the EIS should delay its completion until at least three months after completion of these studies.	5	The electrometallurgical treatment demonstration project is scheduled to conclude in August of 1999. At that time DOE will know if it has met the success criteria established by the National Research Council for the electrometallurgical treatment demonstration. Publication of the final report on the electrometallurgical treatment demonstration by the National Research Council may require a few months past the end of the demonstration project. DOE expects that the report will be available before it makes a decision on the management of the sodium-bonded spent nuclear fuel. DOE has prepared a nonproliferation impacts assessment report that addresses the treatment of sodium-bonded spent nuclear fuel.
This EIS is premature. The Draft SBSNF EIS should not be issued for public comment before publication of the National Academy of Science's Independent Assessment Final Report on the demonstration project, which probably won't be issued until October or November 1999. The National Academy of Sciences Final Report is answering the question, "Will it work," not, "Will it help?"	6	DOE believes that the results from the demonstration and the need to effectively utilize available resources justify the preparation of the EIS in parallel with the final demonstration reviews. The National Research Council has conducted ongoing reviews and issued status reports on the Demonstration Project. These reports are available for review and the final report will be considered when a record of decision is formulated.
DOE is premature in preparing this EIS because the demonstration project will not be completed until after the Draft EIS is published.	11	The electrometallurgical treatment demonstration project that began in June of 1996 is scheduled to conclude in August of 1999. At that time DOE will know if it has met the success criteria established by the National Research Council for the electrometallurgical treatment demonstration. DOE has obtained encouraging data from the demonstration to date, and is confident that the technology holds promise for the management of its sodium-bonded spent nuclear fuel inventory. Publication of the final report on the electrometallurgical treatment demonstration by the National Research Council may require a few months past the end of the demonstration project. DOE plans to make its decision in January of 2000 based on the NRC final report, and other factors such as cost, environmental consequences, and nonproliferation impacts.
DOE's willingness to proceed at this pace without even the completion of their demonstration project indicates the decision on pyroprocessing was made years ago.	2	DOE has made no decision on how the sodium-bonded spent nuclear fuel should be treated. The EIS addresses reasonable alternatives for treatment of this fuel.
More research and development should be completed before the Record of Decision on the alternatives.	1	DOE believes that enough is known about the alternatives to assess their environmental consequences in the EIS. DOE plans to make its decision on how to manage its sodium-bonded spent nuclear fuel in January 2000 based on such factors as technical feasibility, cost, environmental consequences, and nonproliferation impacts.

Issues	*No. of Comments*	*DOE Responses*
The EIS is premature in that there has not been enough time allowed to include the cost analysis.	1	A report comparing costs of the alternatives will be made available to the public during the public comment period for the Draft EIS.
We question the issuance of the Notice of Intent at this time and believe that it should be withdrawn pending compilation of all the technical documentation necessary to inform the scoping process.	1	DOE believes that adequate presentations, displays and written materials on the proposed action and alternatives were provided to the public during the scoping process.
Although there is a regulatory driver for removal of this fuel from Idaho, that is not until 2035, and budget maintenance does not justify going ahead with this process until concerns about its technical feasibility, cost-effectiveness, and potential for proliferation have been adequately addressed. I recommend that DOE provide compelling evidence that it is prudent to proceed with preparing an EIS at this time.	2	DOE believes that enough is known about the alternatives to assess their environmental consequences in the EIS. DOE plans to make its decision on how to manage its sodium-bonded spent nuclear fuel in January 2000 based on such factors as technical feasibility, cost, environmental consequences, and nonproliferation impacts.
Miscellaneous		
This activity could be viewed as corporate welfare which, whether true or not, always is a concern.	2	DOE has identified the purpose and need for the proposed action, which is found in Section 1.2 of the Draft EIS. Action is necessary for the responsible management of DOE's inventory of sodium-bonded spent nuclear fuel.
The intent of the agreement between the Governor of Idaho and DOE involves removing large amounts of radioactive materials, not just spent nuclear fuel.	1	The approximate 60 tons of sodium-bonded spent nuclear fuel currently stored in Idaho contains radioactive materials that cannot be reused, recycled, or disposed of in its current condition. Part of the intent of DOE's proposal is to prepare these materials for disposal or possible reuse for commercial purposes.
If a source is referenced in the EIS, it should be summarized in the EIS (e.g., EAR in the Depleted Uranium Hexafluoride Programmatic EIS).	1	Some reference documents are very large and difficult to summarize. Where practical, DOE has provided a brief summary of reference documents in the EIS.
DOE is not going to consider public comments; instead it is engaging in a show process that meets the bare minimum legal requirements.	1	DOE is considering and will continue to consider public comments in its sodium-bonded spent nuclear fuel management decision process. For example, DOE will provide a comparative cost report and a nonproliferation impacts report to the public in response to comments received during the scoping process. Further, DOE has reformulated its proposed action in response to public comments.
It seems a bit of a waste of the public's time to continue to have these EISs in which we comment saying, "Slow down, we want more information," and DOE says, "Sure," and proceeds right along with its decision in the first place.	1	DOE is committed to providing the public the opportunity to review and comment on the proposed action to manage its inventory of sodium-bonded spent nuclear fuel.
This is not an EIS asking, "We've got a bunch of sodium-contaminated fuel. What should we do with it? We have the following five alternatives." We don't have an action that says, "We need to treat this fuel. We have EISs on it. We want to do pyroprocessing." It is lip service to the other alternatives that are available to deal with this spent nuclear fuel.	1	In response to public comments, DOE has revised the proposed action of the EIS 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.

Issues	*No. of Comments*	*DOE Responses*
We are gravely concerned with the project. We oppose it. We have opposed it all along.	1	DOE acknowledges the commentor's opposition to the proposed action.
That DOE is not waiting for the National Academy of Sciences' Final Report raises a question that Pit Nine also raises. DOE gets a lot of research and development money every year; do the data you collect mean anything?	1	The electrometallurgical treatment demonstration project that began in June of 1996 is scheduled to conclude in August of 1999. At that time DOE will know if it has met the success criteria established by the National Research Council for the electrometallurgical treatment demonstration. DOE has obtained encouraging data from the demonstration to date, and is confident that the technology holds promise for the management of its sodium-bonded spent nuclear fuel inventory. Publication of the final report on the electrometallurgical treatment demonstration by the National Research Council may require a few months past the end of the demonstration project. DOE plans to make its decision in January of 2000 based on the NRC final report, and other factors such as cost, environmental consequences, and nonproliferation impacts.
What is the endpoint for the National Research Council's waste characterization study? Is it a moving target or a dead horse?	1	The National Research Council is reviewing the waste qualification process and the acceptability of the waste forms.
I would like to see the products identified [cost analysis, nonproliferation analysis] in the briefing placed on a schedule that fits into the Secretary of Energy's decision on the Record of Decision. This schedule ought to be made available to the stakeholders.	1	DOE is preparing a nonproliferation impacts assessment report that addresses the treatment of sodium-bonded spent nuclear fuel. This report will be made available to the public during the Draft EIS public comment period. DOE is also preparing a comparative cost report which will be made available to the public during the Draft EIS public comment period.
In the past, DOE has had to redo work because of an inadequate initial assessment of a problem. The commentor hopes DOE will avoid such costly problems by proceeding only if it is clear that treatment is necessary. The commentor will be pleased to see DOE proceed with treating the spent nuclear fuel once adequate environmental documentation has been completed and once it has been established that treatment will be necessary before disposal.	1	This NEPA process will aid DOE in making an informed decision on how to proceed with the management of its sodium-bonded spent nuclear fuel. The alternatives analyzed in this EIS include no action and direct disposal with no treatment. DOE will make its decision in January of 2000 based on the analytical results of this EIS combined with public comments on the Draft EIS and the outcome of the demonstration project, as well as cast, schedule, and nonproliferation considerations.
Would it not be more realistic to base risk analysis on a Hormissis theory rather than the Linear Threshold theory?	1	The EIS acknowledges that there are other views on the effects of radiation at low dose rates. However, the linear dose response is the most accepted as well as the most conservative of current models, and is therefore appropriate for this analysis.
Press for the quickest, most scientifically proven solution to the preparation of this spent nuclear fuel for a repository.	1	DOE will make its decision in January 2000 based on the analytical results of this EIS combined with public comments on the Draft EIS and the outcome of the demonstration project, as well as cost, schedule, and nonproliferation considerations.
Has integration/consolidation with other treatment/conditioning being performed at other DOE sites (Hanford, Savannah River) been considered?	1	DOE has considered the use of other DOE facilities as options for the management of sodium-bonded spent nuclear fuel. These issues were a major consideration of the DOE Programmatic Spent Nuclear Fuel EIS (April 1995). Alternatives 3 and 5 of the current EIS involve the use of two different facilities at the Savannah River Site in South Carolina.

Issues	*No. of Comments*	*DOE Responses*
What happens in the No Action [Alternative] after 2035?	1	Under the No Action Alternative, the EIS evaluates the viability of direct disposal of sodium-bonded spent nuclear fuel in a geologic repository with no treatment, as well as storing the spent nuclear fuel and pursuing the research and development of a new or immature technology
Can the sodium be leached from the uranium?	1	The bond sodium could be melted and drained from the blanket fuel. The melt and drain process would not be effective on the sodium-bonded driver fuel because some of the bond sodium is inside, or encapsulated within the uranium material and the uranium has become mechanically attached to the stainless-steel cladding.
Put the uranium into commercial fuel.	1	Although DOE has not made a decision regarding the disposition of low-enriched uranium, there is a possibility that the low-enriched uranium could be sold to the commercial reactor fuel industry as a feedstock material.
Few details about the [electrometallurgical treatment] process were provided [in the presentation].	1	The intent of the public scoping meeting presentation was to give the public a general overview of the NEPA process, the preferred alternative (electrometallurgical treatment), and other alternatives. The public meeting presentations during the Draft EIS comment period will contain more detail about the electrometallurgical treatment process.
We believe that important questions about cost and waste characterization have been left out of most reviews of this program and urge the Energy Information Agency take an honest, comprehensive look at these issues.	1	As requested by members of the public during the scoping process, DOE is preparing a comparative cost report which will be made available to the public during the Draft EIS comment period. DOE will make its decision in January of 2000 based on the outcome of the Electrometallurgical Treatment Demonstration Project, and other factors such as cost, environmental consequences, and nonproliferation impacts.
This program was featured on *NBC Nightly News* as a "Fleecing of America." According to DOE, this program is being created to cover the "redirection of valuable intellectual and physical resources at ANL.....as a result of the shutdown of the nuclear breeder reactor program known as the Advanced Liquid Metal Reactor). We are outraged that a key piece of a program that was supposedly terminated by Congress—the Advanced Liquid Metal Reactor—continues to squander taxpayer dollars on questionable "termination costs" and a wrong-minded "redirection" program known as pyroprocessing or electrometallurgical treatment at ANL. ...We are extremely concerned that this new "Nuclear Technology Research and Development" program represents nothing more than a continuation of the fuel reprocessing activities supported by the Advanced Liquid Metal Reactor program	1	The electrometallurgical treatment technology under consideration in the EIS for treating sodium-bonded spent nuclear fuel is a technology that was originally developed as part of DOE's Advanced Liquid Metal Reactor Program, which was discontinued in 1994. This technology was developed at significant expense to the taxpayer. DOE would be remiss in its responsibilities not to evaluate the potential application of this technology to the Department's sodium-bonded spent nuclear fuel. DOE believes that its proposal to apply electrometallurgical technology to the management of its sodium-bonded spent nuclear fuel inventory has the potential to solve a significant problem for the Nation.
DOE's record with other reprocessing technologies has been abysmal.	1	DOE has successfully used reprocessing technologies in the past to provide nuclear materials for research and defense purposes. The use of PUREX processing for the declad and cleaned blanket fuel [Alternative 3] is a viable option..

Issues	***No. of Comments***	***DOE Responses***
The [Snake River] Alliance encourages DOE to include ANL-W as part of INEEL in environmental analyses.	1	DOE has included the ANL-W facility as part of the INEEL in analyzing the environmental consequences of the alternatives in this EIS as well as the *DOE Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*.
The commentor would prefer to see the spent nuclear fuel treated only once if possible.	1	DOE also would prefer to treat its sodium-bonded spent nuclear fuel only once, if at all, before its final disposition.
To support informed public review of the Draft EIS, the schedule for this EIS should allow for adequate public review of related documents before the close of the public comment period.	1	The schedule for this EIS allows 45 days for public comment, in accordance with NEPA requirements. Related reports such as those on costs and nonproliferation issues will be available to the public within the same time frame as this Draft EIS.

APPENDIX B
IMPACT ASSESSMENT METHODS

B.1 INTRODUCTION

This appendix briefly describes the methods used to assess the potential direct, indirect, and cumulative effects of the treatment and management of sodium-bonded spent nuclear fuel. Included are impact assessment methods for air quality; water resources; socioeconomics; waste management; and cumulative impacts. Each section is organized so that the affected resource is described first, and then the impact assessment method is presented. Methodologies were not developed for land resources; site infrastructure; noise; geology and soils; ecological resources; and cultural and paleontological resources, since impacts to these resources either would not occur or would be very small. This is because new construction would not be required, airborne and aqueous effluents would be controlled and permitted, and infrastructure requirements would not change for any of the treatment and management alternatives. Descriptions of the methods for the evaluation of human health effects from normal operations; facility accidents; transportation; and environmental justice are presented in Appendices E, F, G, and H, respectively.

Impact analysis varied with the resource area. For air quality, for example, estimated pollutant concentrations from the proposed facilities were compared with the appropriate regulatory standards or guidelines. Comparison with regulatory standards is a commonly used method for benchmarking environmental impacts and was done here to provide perspective on the magnitude of the identified impacts. The analysis of waste management impacts compared waste generated by the management of sodium-bonded spent nuclear fuel to the capacities of waste management facilities. Impacts in all resource areas were analyzed consistently; that is, the impact values were estimated using a consistent set of input variables. Also, similar presentations were developed to facilitate the comparison of alternatives.

B.2 AIR QUALITY

B.2.1 Description of Affected Resources

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. For purposes of this environmental impact statement (EIS), only outdoor air pollutants were addressed. They may be in the form of solid particles, liquid droplets, gases, or a combination of these forms. Generally, they can be categorized as primary pollutants (those emitted directly from identifiable sources) and secondary pollutants (those produced in the air by interaction between two or more primary pollutants, or by reaction with normal atmospheric constituents that may be influenced by sunlight). Air pollutants are transported, dispersed, or concentrated by meteorological and topographical conditions. Thus, air pollutant emission characteristics, meteorology, and topography affect air quality.

Ambient air quality in a given location can be described by comparing the concentrations of various pollutants in the atmosphere with the appropriate standards. Ambient air quality standards have been established by Federal and state agencies, allowing an adequate margin of safety for protection of public health and welfare from the adverse effects of pollutants in the ambient air. Pollutant concentrations higher than the corresponding standards are considered unhealthy; those below such standards are considered acceptable.

The pollutants of concern are primarily those for which Federal and state ambient air quality standards have been established, including criteria air pollutants, hazardous air pollutants, and other toxic air compounds. Criteria air pollutants are those listed in 40 CFR 50. Hazardous air pollutants and other toxic compounds are those listed in Title I of the 1990 Clean Air Act, as amended; those regulated by the National Emissions Standards for Hazardous Air Pollutants; and those that have been proposed or adopted for regulation by the respective state or are listed in state guidelines. Also of concern are air pollutant emissions that may contribute to the depletion of stratospheric ozone or global warming.

Areas with air quality better than the National Ambient Air Quality Standards (NAAQS) for criteria air pollutants are designated as being in attainment, while areas with air quality worse than the NAAQS for such pollutants are designated as being in nonattainment. Areas may be designated as unclassified when sufficient data for attainment status designation are lacking. Attainment status designations are assigned by county, metropolitan statistical area, consolidated metropolitan statistical area, or portions thereof. Air Quality Control Regions designated by the U.S. Environmental Protection Agency (EPA) are listed in 40 CFR 81.

For locations that are in an attainment area for criteria air pollutants, prevention of significant deterioration regulations limit pollutant emissions from new sources and establish allowable increments of pollutant concentrations. Three prevention of significant deterioration classifications are specified with the criteria established in the Clean Air Act amendments. Class I areas include national wilderness areas; memorial parks larger than 2,020 hectares (5,000 acres); national parks larger than 2,430 hectares (6,000 acres); and areas that have been redesignated as Class I. Class II areas are all areas not designated as Class I. No Class III areas have been designated. Idaho National Engineering and Environmental Laboratory (INEEL) and the Savannah River Site (SRS) are within attainment areas (Class II) for the criteria air pollutants. INEEL is located about 50 kilometers (33 miles) from the Craters of the Moon National Monument Class I area. There are no Class I areas within 100 kilometers (62 miles) of SRS.

Baseline air quality is typically described in terms of pollutant concentrations modeled for existing sources at each site and background air pollutant concentrations measured near the sites. For this analysis, concentrations for existing sources were obtained from the sites (Moor and Peterson 1999) and from the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement* (DOE 1998). These concentrations were compared with Federal and state regulations or limits (**Table B–1**). To determine human health risk, modeled chemical concentrations in air were weighed against chemical-specific toxicity values.

B.2.2 Description of Impact Assessment

Potential air quality impacts of pollutant emissions were evaluated for each alternative. This assessment included a comparison of emissions from each alternative with applicable Federal and state ambient air quality standards. If both Federal and state standards exist for a given pollutant and averaging period, compliance was evaluated using the more stringent standard.

Table B–1 Impact Assessment Protocol for Air Quality

Resources	*Required Data*		*Measure of Impact*
	Affected Environment	*Alternative*	
Criteria air pollutants and other regulated pollutants[a]	Modeled ambient concentrations (micrograms per cubic meter) of air pollutants from existing sources at site	Emission rate (kilograms per year) of air pollutants from facility and concentrations of air pollutants	Contribution of proposed alternative and total concentration of each pollutant at or beyond site boundary compared to applicable standard
Toxic/hazardous air pollutants[b]		Emission rate (kilograms per year) of toxic air pollutants from facility (micrograms per cubic meter)	

[a] Carbon monoxide; hydrogen fluoride; lead; nitrogen oxides; ozone; particulate matter with an aerodynamic diameter less than or equal to 10 microns; particulate matter with an aerodynamic diameter less than 2.5 microns; sulfur dioxide; total suspended particulates.

[b] Clean Air Act Title III pollutants, pollutants regulated under the National Emission Standards for Hazardous Air Pollutants, and other state-regulated pollutants.

Air pollutant emissions and concentrations data for each alternative, including the No Action Alternative, were based on information obtained in response to data requests to INEEL (ANL 1999, Moor and Peterson 1999) and on the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement* (DOE 1998). INEEL emissions and corresponding concentrations were not quantified, but are expected to be well below regulatory concern. For SRS, concentrations were obtained by scaling the SRS Spent Nuclear Fuel Draft EIS concentrations based on the mass of blanket fuel to be processed under each alternative.

Ozone is typically formed as a secondary pollutant in the ambient air (troposphere). It is formed from primary pollutants such as nitrogen oxides and volatile organic compounds, which emanate from vehicular (mobile), natural, and other stationary sources, mixing in the presence of sunlight. Ozone is not emitted directly as a pollutant from the sites. Although ozone may be regarded as a regional issue, specific ozone precursors, notably nitrogen dioxide and volatile organic compounds, were analyzed as applicable to the alternatives under consideration.

Emissions of potential stratospheric ozone-depleting compounds such as chlorofluorocarbons were not evaluated, as no emissions of these pollutants were identified.

B.3 WATER RESOURCES

B.3.1 Description of Affected Resources

Water resources are the surface and subsurface waters that are suitable for human consumption; agricultural purposes; irrigation; or industrial/commercial purposes, and that could be impacted by the treatment of sodium-bonded spent nuclear fuel. This analysis involves the review of engineering estimates of expected water use and effluent discharges associated with the alternatives addressed in this EIS, and the impacts of these alternatives on local water quality (including surface water and groundwater).

Surface water flow data and water quality data were obtained from existing reports. Groundwater users, information on water use rights, and groundwater quality data also were obtained from existing reports.

B.3.2 Description of Impact Assessment

B.3.2.1 Water Use

The assessment of alternatives analyzed how the volume of current water usage and effluent discharges would change as a result of each alternative addressed in this EIS. Determination of the impacts of the alternatives on water usage and effluent discharge is summarized in **Table B–2**.

Table B–2 Impact Assessment Protocol for Water Use and Effluent Discharge

Resources	*Required Data*		*Measure of Impact*
	Affected Environment	*Alternative*	
Surface water availability	Surface waters near the facilities, including average flow and numbers of downstream users	Volumes of withdrawals from and discharges to surface waters	Changes in availability to downstream users of water for human consumption, irrigation, or animal feeding[a]
Groundwater availability	Groundwater near the facilities, including existing water rights for major water users, and contractual agreements for water supply use within impacted area	Volume of withdrawals from groundwater	Changes in availability of groundwater for human consumption, irrigation, or animal feeding

[a] For surface water availability, an impact is assumed if withdrawals exceed 10 percent of the 7-day, 10-year low-flow of the stream.

If the determination reflected an increase in water use or effluent discharge, then an evaluation of the design capacity of the water and effluent treatment facilities was made to determine whether the design capacity would be exceeded by the additional flow. If the combined flow (i.e., the existing flow plus that of the proposed activities) was less than the design capacity of the water and effluent treatment plants, then it was assumed that there would be no impact on water availability for local users, nor on the receiving stream from effluent discharges. Since flows from the facilities proposed to treat sodium-bonded spent nuclear fuel were found to not exceed the design capacity of the existing water or effluent treatment facilities, no additional analysis of water availability was performed.

B.3.2.2 Water Quality

The water quality impact assessment for this EIS analyzed how effluent discharges to surface water and groundwater resulting from the alternatives would affect current water quality. The determination of the impacts of the alternatives is summarized in **Table B–3**, and consisted of a comparison of the projected water quality with relevant regulatory standards such as the Clean Water Act, Safe Drinking Water Act, state regulations, and existing permit conditions. Separate analyses were conducted for surface water and groundwater impacts, as described below.

Table B–3 Impact Assessment Protocol for Water Quality

Resources	*Required Data*		*Measure of Impact*
	Affected Environment	*Alternative*	
Surface water quality	Surface waters near the facilities in terms of stream classifications and changes in water quality	Expected contaminants and contaminant concentrations in discharges to surface water	Compliance of discharges to surface water with relevant standards of Clean Water Act or with state regulations and existing National Pollutant Discharge Elimination System (NPDES) permits
Groundwater quality	Groundwater near the facilities in terms of classification, presence of designated sole–source aquifers, and changes in quality of groundwater	Expected contaminants and contaminant concentrations in discharges that could reach groundwater	Concentrations of contaminants in groundwater exceeding standards established in accordance with Safe Drinking Water Act or state regulations

Surface Water Quality

The evaluation of surface water quality impacts focused on the quality and quantity of effluent to be discharged and the quality of the receiving stream upstream and downstream from the discharge. The evaluation of effluent quality featured review of the expected parameters, such as the design average, as well as the effluent parameters reflected in the existing or expected NPDES permit. Those parameters include metals; organic and inorganic chemicals; radionuclides; and any other parameters that affect the local environment. Water quality management practices were reviewed to ensure that NPDES permit limitations would be met. Factors that currently degrade water quality were also identified.

Groundwater Quality

No effluent discharges to groundwater are anticipated from any of the alternatives. Therefore, an analysis of impacts to groundwater quality was not performed.

B.4 SOCIOECONOMICS

B.4.1 Description of Affected Resources

Socioeconomic impacts are defined in terms of changes to the demographic and economic characteristics of a region. The number of jobs created by treatment of sodium-bonded spent nuclear fuel could affect regional employment, income, and expenditures. Job creation is characterized by two types: construction jobs related to modification of existing facilities, which may be transient in nature and short in duration, and thus less likely to impact public services; and jobs related to plant operations that are required for a decade or more and, thus, possibly create additional service requirements in the region of influence.

The socioeconomic environment is made up of two geographic regions, the regional economic area and the region of influence. Regional economic areas are made up of regional economies and include industrial and service sector characteristics and their linkages to the communities within a region. These linkages determine the nature and magnitude of any effect associated with a change in regional economic activity. For example, as work expands within a region, the money spent on accomplishing this work flows into the local economy, where it is spent on additional jobs, goods, and services within the regional economic area.

Similarly, potential demographic impacts were assessed for the region of influence. The region of influence could represent a smaller geographic area—one in which only the housing market and local community services would be significantly affected by a given alternative. Site-specific regions of influence were

identified as those counties in which at least 90 percent of the site's work force reside. This distribution reflects an existing residential preference for people currently employed at the sites, and was used to estimate the distribution of new workers supporting the alternatives.

B.4.2 Description of Impact Assessment

The socioeconomic impact assessment analyzes both the potential positive and negative impacts of each alternative, including the No Action Alternative. For each regional economic area, data were compiled on the current socioeconomic conditions, including unemployment rates, economic industrial and service sector activities, and the civilian labor force. Work force and cost requirements of each alternative were determined to measure their possible effect on these socioeconomic conditions. For each region of influence, census statistics were compiled on population, housing demand, and community services. U.S. Census Bureau population forecasts for the regions of influence were combined with overall projected work force requirements for each of the alternatives being considered at each of the sites to determine the extent of impacts to housing demand and levels of community services (**Table B–4**).

Table B–4 Impact Assessment Protocol for Socioeconomics

Resources	*Required Data*		*Measure of Impact*
	Affected Environment	*Alternative*	
Regional Economic Characteristics			
Work force requirements	Site work force projections from DOE sites	Estimated construction and operating staff requirements and schedule	Work force requirements added to sites' work force projections
Regional economic area civilian labor force	Labor force projections based on state population projections		Work force requirements as a percentage of the civilian labor force
Unemployment rate	1996 unemployment rates in counties surrounding sites and in host states		Projected change in unemployment rates
Population and Housing			
Population	Latest available population projection estimates from the U.S. Census Bureau	Estimated contribution to projected population	Projected change in population projection
Housing (percent of occupied housing units)	Latest available rates from the U.S. Census Bureau	Assess potential need for housing units to meet work force requirements	Impacts are not expected since work force requirements are small
Community Services			
Education Percent operating capacity for school districts in region of influence	Latest available rates from the U.S. Census Bureau	Assess potential need for new schools	Impacts are not expected since work force requirements are small
Teacher-to-student ratio		Assess potential need for additional teachers	
Public safety Ratio of police and firefighters to 100,000 residents		Assess potential need for new officers	
Health care Number of hospital beds and physicians per 100,000 residents		Assess potential need for hospitals and physicians	

B.5 WASTE MANAGEMENT

B.5.1 Description of Affected Resources

The operation of support facilities for treating sodium-bonded spent nuclear fuel would generate several types of waste, depending on the alternative. Such wastes include the following:

- **High-level:** The highly radioactive waste material that results from the processing of spent nuclear fuel, including liquid waste produced directly in processing, and any solid waste derived from the liquid. High-level radioactive waste contains transuranic waste and fission products in combinations requiring permanent isolation.

- **Transuranic:** Waste containing more than 100 nanocuries of alpha-emitting transuranic isotopes per gram of waste with half-lives greater than 20 years, except for: (1) high- level radioactive waste; (2) waste that DOE has determined, with the concurrence of the EPA, does not need the degree of isolation required by 40 CFR 191; and (3) waste that the U.S. Nuclear Regulatory Commission (NRC) has approved for disposal, case by case, in accordance with 10 CFR 61. Mixed transuranic waste contains hazardous components regulated under the Resource Conservation and Recovery Act (RCRA).

- **Low-level:** Waste that contains radioactivity and is not classified as high-level radioactive waste; transuranic waste; spent nuclear fuel; or the tailings or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material. Test specimens of fissionable material irradiated for research and development only, and not for the production of power or plutonium, may be classified as low-level radioactive waste, provided the transuranic concentration is less than 100 nanocuries per gram of waste.

- **Mixed:** Low-level radioactive waste that also contains hazardous components regulated under RCRA.

- **Hazardous:** Under RCRA, a waste that, because of its characteristics, may (1) cause or significantly contribute to an increase in mortality or an increase in serious, irreversible, or incapacitating reversible illness; or (2) pose a substantial present or potential hazard to human health or the environment when improperly treated, stored, transported, disposed of, or otherwise managed. Hazardous wastes appear on special EPA lists or possess at least one of the following characteristics: ignitability, corrosivity, reactivity, or toxicity. This category does not include source, special nuclear, or byproduct material as defined by the Atomic Energy Act.

- **Nonhazardous:** Discarded material including solid, liquid, semisolid, or contained gaseous material resulting from industrial, commercial, mining, and agricultural operations and from community activities. This category does not include source, special nuclear, or byproduct material as defined by the Atomic Energy Act.

- **Other Wastes:** Miscellaneous waste streams such as fuel assembly hardware, metal and ceramic waste forms, and spent processing chemicals.

Wastes associated with the alternatives for treating the sodium-bonded spent nuclear fuel would be managed in existing or already-planned-for treatment, storage, and disposal facilities. The management could have an impact on existing site facilities. Wastes generated during modifications to existing facilities could produce additional hazardous debris.

Waste management activities in support of treatment of sodium-bonded spent nuclear fuel would be contingent on Records of Decision issued for the *Final Waste Management Programmatic Environmental Impact*

Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste (DOE 1997a). Depending on future waste type-specific Records of Decision, in accordance with that EIS, wastes could be treated and disposed of on site or at regionally or centrally located waste management centers. According to the Transuranic Waste Record of Decision issued January 20, 1998, transuranic and transuranic mixed waste would be treated on site according to current planning-basis Waste Isolation Pilot Plant waste acceptance criteria and shipped to the Waste Isolation Pilot Plant for disposal. The impacts of disposing of transuranic waste at the Waste Isolation Pilot Plant are described in the *Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement* (DOE 1997b). Per the Hazardous Waste Record of Decision issued August 5, 1998, nonwastewater hazardous waste would continue to be treated and disposed of at offsite commercial facilities, with SRS continuing to treat some of its own hazardous waste on site in existing facilities, where this is economically favorable.

B.5.2 Description of Impact Assessment

As shown in **Table B–5**, impacts were assessed by comparing the projected waste stream volumes generated from the alternatives at each site with current site waste generation rates and storage volumes. For sodium-bonded spent nuclear fuel treatment, only the impacts related to the capacities of waste management facilities were considered. Environmental impacts of waste management facility operation are evaluated in other facility-specific or site-wide National Environmental Policy Act (NEPA) documents. Projected waste generation rates for the alternatives were compared with processing rates and capacities of those existing treatment, storage, and disposal facilities likely to be involved in managing the additional waste.

The waste generation rates associated with sodium-bonded spent nuclear fuel treatment were either provided by the sites' technical personnel or estimated based on evaluating similar processes, with adjustments made to account for differences in the amounts of materials processed.

Table B–5 Impact Assessment Protocol for Waste Management

Resources	*Required Data*		*Measure of Impact*
	Affected Environment	*Alternative*	
Waste management capacity related to: High-level radioactive waste Transuranic waste Low-level radioactive waste Mixed waste Hazardous waste Nonhazardous waste Other wastes	Site generation rates (cubic meters per year) for each waste type Site management capacities (cubic meters) or rates (cubic meters per year) for potentially affected treatment, storage, and disposal facilities for each waste type	Generation rates (cubic meters per year) of each waste type from modification and operation of existing facilities used to treat the sodium-bonded spent nuclear fuel	Combination of waste generation volumes from: (1) facilities that treat sodium-bonded spent nuclear fuel, and (2) current site and additional future generation volumes, in comparison to the capacities of applicable waste management facilities
Disposal capacity for transuranic waste (including mixed transuranic waste)[a]	Transuranic waste volume (cubic meters) expected to be disposed of at the Waste Isolation Pilot Plant Capacity at the Waste Isolation Pilot Plant (cubic meters)	Total transuranic waste generated (cubic meters) by spent nuclear fuel treatment facilities	Combination of transuranic waste generation volumes from: (1) facilities that treat sodium-bonded spent nuclear fuel, and (2) current site transuranic waste generation volume, in comparison to the capacity of the Waste Isolation Pilot Plant

[a] This additional entry is made for transuranic waste disposal because of its comparison with Waste Isolation Pilot Plant capacity.

B.6 CUMULATIVE IMPACTS

Cumulative impacts can result from individually minor, but collectively significant actions taking place over a period of time (40 CFR 1508.7). The cumulative impact analysis for this EIS involved combining the impacts of the sodium-bonded spent nuclear fuel treatment alternatives (including No Action) with the impacts of other past, present, and reasonably foreseeable activities in a region of influence.

The regions of influence for different resources can vary widely in extent. For example, the region of influence for waste management would generally be confined to the site itself, whereas the region of influence for human health would include areas extending out to 80 kilometers (50 miles) from each site.

In general, cumulative impacts were calculated by adding the values for the baseline affected environment (i.e., conditions attributable to past and present actions by DOE and other public and private entities), the proposed action, and future actions. This cumulative value was then weighed against the appropriate impact indicators to determine the potential for impact. For this cumulative impact assessment, it was conservatively assumed that all facilities would operate concurrently at the DOE sites. Only selected indicators of cumulative impacts (**Table B–6**) were evaluated.

Table B–6 Selected Indicators of Cumulative Impacts

Category	*Indicator*
Resource use	Electricity use Water use Workers required
Air quality	Percent of NAAQS for criteria pollutants
Human health	Public • Offsite population • Maximally exposed individual dose • Total dose • Fatalities • Workers • Average dose • Total dose • Fatalities
Waste and spent nuclear fuel	Site waste generation rate versus capacity for: High-level radioactive waste Transuranic waste Low-level radioactive waste Mixed waste Hazardous waste Sanitary wastewater

The analysis focused on the potential for cumulative impacts at each candidate site from DOE actions under detailed consideration at the time of this EIS (**Table B–7**). Non-DOE actions were also considered where information was readily available. Public documents prepared by agencies of Federal, state, and local governments were the primary sources of information for non-DOE actions.

Table B–7 Other Past, Present, and Reasonably Foreseeable Actions Included in the Cumulative Impact Assessments

Activities	*INEEL*	*SRS*
Storage and disposition of weapons-usable fissile materials	X	X
Disposition of surplus highly enriched uranium		X
Interim management of nuclear materials at SRS		X
Management of waste at SRS		X
Supply and recycling of tritium		X
Management of waste	X	X
Management of spent nuclear fuel and INEEL environmental restoration and waste management	X	X
Management of foreign research reactor spent nuclear fuel	X	X
Shutdown of the river water system at SRS		X
Radioactive releases from the Vogtle Nuclear Power Plant		X
Management of plutonium residues and scrub alloy at Rocky Flats		X
Stewardship and management of the nuclear weapons stockpile		X
Accelerator production of tritium at SRS		X
Construction and operation of a tritium extraction facility at SRS		X

It was assumed that construction impacts related to internal modification of existing facilities would not be cumulative, because construction typically is short in duration and construction impacts generally are temporary. Decontamination and decommissioning of the facilities utilized for the treatment of sodium-bonded spent nuclear fuel was not addressed in the cumulative impact estimates. Given the uncertainty regarding the timing of decontamination and decommissioning, and the fact that facilities could be used for other projects, any impact estimate at this time would be premature. A detailed decontamination and decommissioning evaluation will be provided in follow-on NEPA documentation closer to the actual time of those actions.

Recent site-wide NEPA documents (**Table B–8**) provide the latest comprehensive evaluation of cumulative impacts for the sites.

Table B–8 Recent Comprehensive NEPA Documents for DOE Sites Assessed in This EIS

Site	*Document*	*Year*	*Record of Decision First Issued*
INEEL	DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final EIS (DOE 1995a)	1995	March 1996
SRS	SRS Waste Management Final EIS (DOE 1995b)	1995	October 1995

B.7 REFERENCES

ANL (Argonne National Laboratory), 1999, *Response to Data Call from SAIC for Sodium-Bonded Spent Nuclear Fuel Treatment Technologies*, Idaho National Engineering and Environmental Laboratory, Idaho Falls, June.

DOE (U.S. Department of Energy), 1995a, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, DOE/EIS-0203-F, Office of Environmental Management, Idaho Operations Office, Idaho Falls, Idaho, April.

DOE (U.S. Department of Energy), 1995b, *Savannah River Site Waste Management Final Environmental Impact Statement*, DOE/EIS-0217, Savannah River Operations Office, Aiken, South Carolina, July.

DOE (U.S. Department of Energy), 1997a, *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste*, DOE/EIS-0200-F, Office of Environmental Management, Washington, DC, May.

DOE (U.S. Department of Energy), 1997b, *Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement*, DOE/EIS-0026-S-2, Carlsbad Area Office, Carlsbad, New Mexico, September.

DOE (U.S. Department of Energy), 1998, *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement*, DOE/EIS-0279D, Savannah River Operations Office, Aiken, South Carolina, December.

Moor, K. S., and H. K. Peterson, 1999, *INEEL Affected Environment: Supplemental Data Report in Support of the Preparation of the Plutonium-238 Production at ATR Environmental Impact Statement* (Predecisional Draft), INEL/EXT-99-Draft, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho, February.

APPENDIX C
TECHNOLOGY DESCRIPTIONS

The technology options that the U.S. Department of Energy (DOE) has considered for processing sodium-bonded spent nuclear fuel are described in this appendix. Each technology is described in the context of processing sodium-bonded spent nuclear fuel driver and/or blanket assemblies. A brief discussion of the technical maturity of each treatment technology is included at the end of each technology description. The technical maturity of the technologies range from mature technologies that have been previously demonstrated by DOE for spent nuclear fuels or in an industrial setting, to immature technologies that have only been demonstrated on a laboratory scale or for which only a conceptual design has been developed.

C.1 ELECTROMETALLURGICAL TREATMENT

The electrometallurgical treatment process for sodium-bonded spent nuclear fuel was developed at the Argonne National Laboratory for processing Experimental Breeder Reactor-II (EBR-II) spent nuclear fuel and blanket assemblies. The process has been demonstrated for the stainless steel-clad uranium alloy fuel used in that reactor. The electrometallurgical treatment process uses electrorefining, an industrial technology used to produce pure metals from impure metal feedstock (DOE 1996). Although most of the sodium-bonded spent nuclear fuel and blanket elements are composed of uranium metal alloys, there are also small quantities of sodium-bonded uranium oxide and uranium carbide fuels. The oxide fuels 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 fuels and blanket elements and 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 dilute acid. This oxide would then be converted to uranium metal by lithium metal in a molten salt solution and processed by the electrometallurgical treatment process with other sodium-bonded spent nuclear fuels and blankets.

The individual steps in the electrometallurgical treatment process are described below. A diagram of the electrometallurgical process is shown in **Figure C–1**.

Disassembly: Although the fuel and blanket assemblies have been mostly disassembled, there may be some assemblies that need to be removed. The assembly hardware would be separated from the fuel elements that contain uranium and fission products by cutting the assemblies and physically separating the fuel elements. The fuel elements would be placed into a container for transfer to a hot cell containing an inert (argon) atmosphere for the remaining treatment steps. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at Argonne National Laboratory-West (ANL-W). This is a normal waste stream for ANL-W operations, and the separated hardware would be handled in accordance with normal site waste management practices.

Fuel Element Chopping: The fuel elements containing sodium-bonded driver fuel or the blanket fuel materials would be placed in a machine for cutting the fuel elements into small pieces. The section of the element containing the fuel and sodium would be sheared into short segments. The section of the element containing the gas space (plenum) would be left intact. This section of the fuel pin cladding and the spacer wire would go into the metal waste stream. The sheared fuel segments would be placed in perforated, stainless steel

baskets to form an anode (positive electrode where oxidation would occur) for the electrorefiner. About 10 kilograms (22 pounds) of uranium would be contained in these baskets of sheared fuel elements.

During shearing of the hollow end (plenum) of the fuel pin, some fission product gases (primarily tritium and krypton) would be released to the argon cell atmosphere. These gases would eventually pass through high efficiency particulate air filters and be released up the emissions stack to the environment. All air emissions would be monitored and recorded.

Electrorefining: The electrorefiner is a machine in which the main electrometallurgical processes would occur. The electrorefiner vessel is made of steel. At its operating temperature of 500 °C (930 °F), the vessel would contain a molten mixture of two salts, lithium chloride and potassium chloride. The electrorefiner also would have two or more electrodes: one or more anodes and one or more cathodes (negative electrodes where reduction would occur). Each anode would have baskets to hold the spent nuclear fuel pieces, and each cathode would consist of a bare steel rod, where uranium metal would be collected.

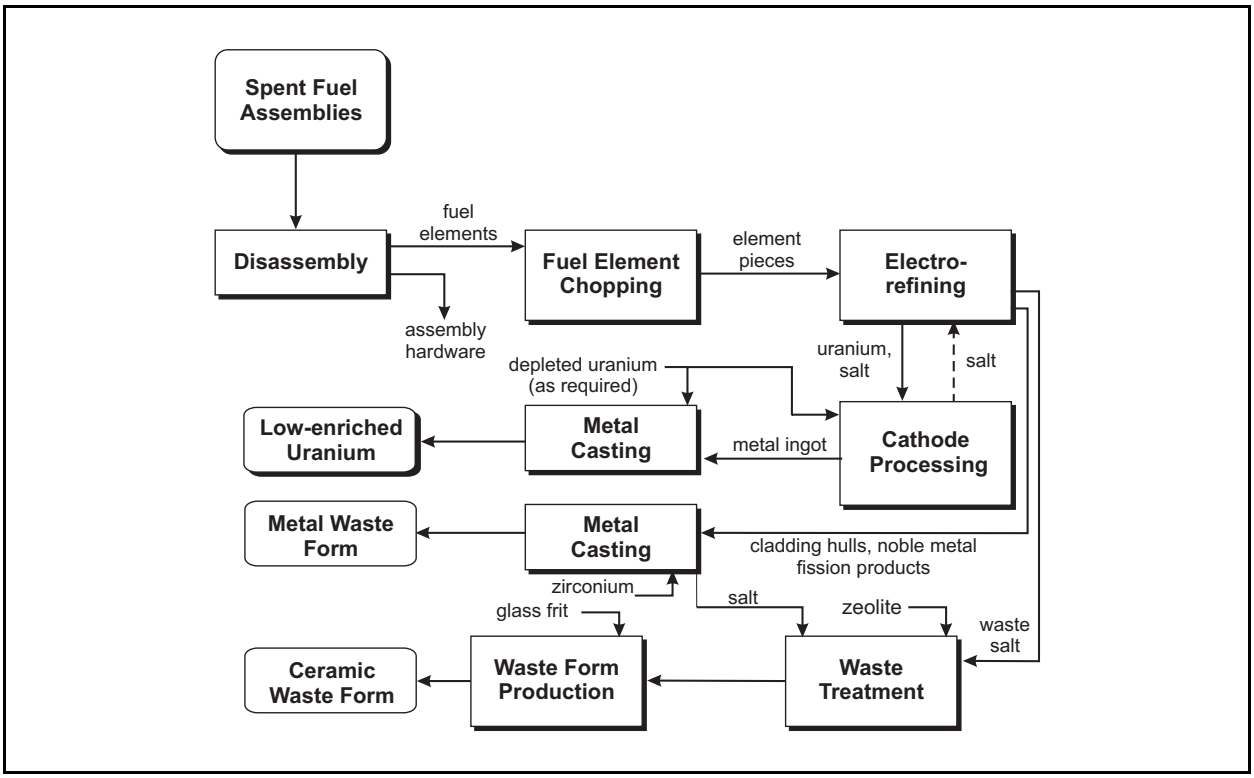

Figure C–1 Electrometallurgical Treatment Process Flow Diagram

The chopped fuel elements would be loaded into anode baskets and then lowered into the molten process salt. Upon application of an electric current between the anodes and cathodes, uranium, plutonium and other transuranic elements, most of the fission products and the sodium would be oxidized and dissolved into the salt. Uranium ions would be deposited at the cathode. Crystalline deposits of uranium would grow for 24 to 72 hours until almost all of the uranium in the anode baskets has been dissolved. The uranium-bearing cathodes would be raised into the gas space in the electrorefiner to allow some of the molten salt to drain away, although salt would adhere to each cathode. Each cathode would then be removed from the electrorefiner. The uranium deposit would be mechanically harvested and stored in the argon cell in a canister until it could be processed in the cathode processor.

The stainless steel cladding hulls and noble metal fission products would remain undissolved in the anode baskets. They would be removed from the electrorefiner and temporarily stored, prior to melting, into metal waste-form ingots. The reactive fission products and transuranic elements would remain in the electrorefiner salt. The sodium would be in the form of sodium chloride (common table salt) as a part of the molten salt mixture.

Electrometallurgical process would use two electrorefining designs: Mark IV (driver fuel) and Mark V (blanket fuel). Mark IV electrorefiner design uses a layer of cadmium to allow recovery of uranium that falls off the cathode during treatment. Mark V design uses a collection basket instead of a cadmium layer.

Cathode Processing: The uranium deposits would be removed from the electrorefiner and treated to remove any adhering salt in the cathode processor, which is a furnace equipped with a vacuum system. The cathode product (along with depleted uranium, in the case of driver fuel) would be heated to about 1,200 °C (2,200 °F), melting both the uranium and the salt. Under vacuum conditions, the salt would distill away from the uranium and condense in a receiver crucible. The uranium would be melted in the cathode processor crucible and then solidified into an interim product ingot, which would be stored before final treatment in a casting furnace.

Uranium Metal Casting: The enriched uranium from driver fuel elements recovered in the electrorefiner would be melted together in a casting furnace with a separate stream of depleted uranium, electromagnetically stirred, and allowed to solidify. In this manner, enriched uranium from the treatment of driver fuel assemblies would be blended with depleted uranium in the casting furnace to form low-enriched, metal ingots. The ingots would then be transferred to the Materials Building within the Zero Power Physics Reactor complex, a controlled storage facility, until a decision is made by DOE regarding final disposition. Similarly, depleted uranium from treatment of blanket assemblies would be melted in a casting furnace and placed into storage until DOE makes a decision on final disposition.

Metal Waste Form: The metal waste form is one of the two high-level radioactive waste forms generated from electrometallurgical treatment of sodium-bonded spent nuclear fuel and blankets. This waste form would consist of metallic ingots used to stabilize the stainless-steel cladding material, non-actinide fuel matrix materials, and noble metal fission products. Actinides that remain in the cladding hulls after dissolution would also be present in the metal waste form. These metals would be melted together in a separate casting furnace from the one used for uranium metal casting. Any salt remaining with the metals would be distilled away under vacuum at about 1,200 °C (2,200 °F). Upon heating to about 1,600 °C (2,200 °F), the metals would melt and form an alloy. A small amount of zirconium metal would also be added to improve performance properties and to produce a lower melting point alloy. After cooling, the metal would solidify into a metallic waste ingot. The typical composition of these ingots would be stainless steel, 15 weight percent zirconium, and about 1 percent noble metal fission products (Goff et al. 1999). These ingots would be stored in interim dry storage at the Radioactive Scrap and Waste Facility pending a decision on disposition.

Treatment of Electrorefiner Wastes: At the end of a processing campaign, fission products and actinides would remain dissolved in the molten salt. The waste salt would be removed from the electrorefiner and allowed to solidify. It then would be crushed and milled to obtain the desired particle size for ceramic waste form production. The liquid cadmium layer at the bottom of the electrorefiner would also be periodically removed, filtered, and returned to the electrorefiner. Filters from this bulk fluid handling system would become part of the metal waste stream.

Ceramic Waste Form Production: The ceramic waste form is the second waste form generated from electrometallurgical treatment of sodium-bonded spent nuclear fuel. The crushed and milled waste salt and dried zeolite would be added to a heated V-mixer. [Zeolites are crystalline aluminosilicates of group I (alkali) and group II (alkaline earth) elements. Their framework is a network of aluminum oxide and silicon oxide

tetrahedra linked by the sharing of oxygen atoms. The networks of tetrahedra in the zeolite form cages in which molecules can be occluded.] The waste salt containing fission products and actinides would be absorbed into the crystal lattice of the zeolite, forming a dry particulate solid. Glass frit (a sand-like material which is glass) then would be mixed with the waste-bearing zeolite and placed in a special metal canister designed to be compressed to a desired and predictable shape. The mixture of material going into the process would be about 75 weight-percent waste-bearing zeolite and 25 weight-percent glass (Goff 1999). This canister would be put into a type of furnace called a hot isostatic press, where it would be subjected to a temperature of 850 °C (1,560 °F) and a pressure of 1,057 kilograms per square centimeter (15,000 pounds per square inch). This would compress the canister and transform the material inside into a single cylinder of glass-bonded zeolite, which is referred to as the ceramic waste form. During compression, the zeolite would be converted to sodalite, a naturally occurring, salt-bearing material. Fission product chlorides would largely remain in the sodalite phase, while actinides (and most of the rare earth elements) would react with residual water in the zeolite to form oxide phases in the waste form. These waste-form cylinders would be packaged and stored in the Radioactive Scrap and Waste Facility until they would be shipped to a geologic repository for disposal.

Technology Maturity: The electrometallurgical treatment process is considered to be a mature technology. DOE demonstrated the process for stainless steel-clad uranium alloy fuel used in the EBR-II reactor. Furthermore, it is an industrial technology used to produce pure metals from impure metal feedstock.

C.2 PLUTONIUM-URANIUM EXTRACTION (PUREX) PROCESS

The PUREX process is a counter-current solvent extraction method used to separate and purify uranium and plutonium from fission product-containing spent nuclear fuel and irradiated uranium targets. DOE has two facilities at the Savannah River Site (SRS), F-Canyon and H-Canyon, that use the PUREX process for the treatment of aluminum-clad fuel and targets. In this EIS, the PUREX process at F-Canyon is being considered for treating declad and cleaned EBR-II and the Fermi-1 blanket fuel. The stainless steel cladding and sodium would be removed from these blanket fuel elements at ANL-W. The cleaned blanket fuel pins would be packaged in aluminum cans and shipped to SRS. The decladding and cleaning activities would be conducted in argon cells at ANL-W facilities. A diagram of the PUREX process is shown in **Figure C–2**.

Disassembly: The first step in the process would be similar to the disassembly process previously described in Section C.1. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at ANL-W and handled in accordance with normal site waste management practices.

Decladding and Sodium Removal: Blanket fuel elements would be brought into an argon-atmosphere hot cell where the ends of the elements would be cut off to expose the sodium within the cladding. The elements then would be cut into segments less than 61 centimeters (24 inches) in length. The fuel element segments would be placed into a crucible and loaded into an induction furnace. The temperature in the furnace would be raised above the melting point of sodium (200 °C [390 °F]) and the molten sodium drained into a collection tank. With most of the sodium removed, the temperature would be raised to 500 °C (930 °F) and a 10^{-4} Torr vacuum would be applied to the chamber. This would volatilize the residual sodium, allowing the sodium vapor to be drawn away from the fuel. The vapor-phase sodium would be condensed in a trap and combined with the sodium, which had been drained into the collection tank, pending further processing. The uranium pins would be mechanically pushed out of the stainless-steel cladding after all the sodium bond had been eliminated. The bare uranium pins would then be packed into aluminum canisters in the Hot Fuel Examination Facility. The canisters, approximately 10 centimeters (4 inches) in diameter and 61 centimeters (24 inches) in length, would be backfilled with an inert gas and sealed. Each canister would contain about 60 kilograms (130 pounds) of depleted uranium fuel pins. The canisters would be placed in a NAC-LWT cask for shipment to SRS.

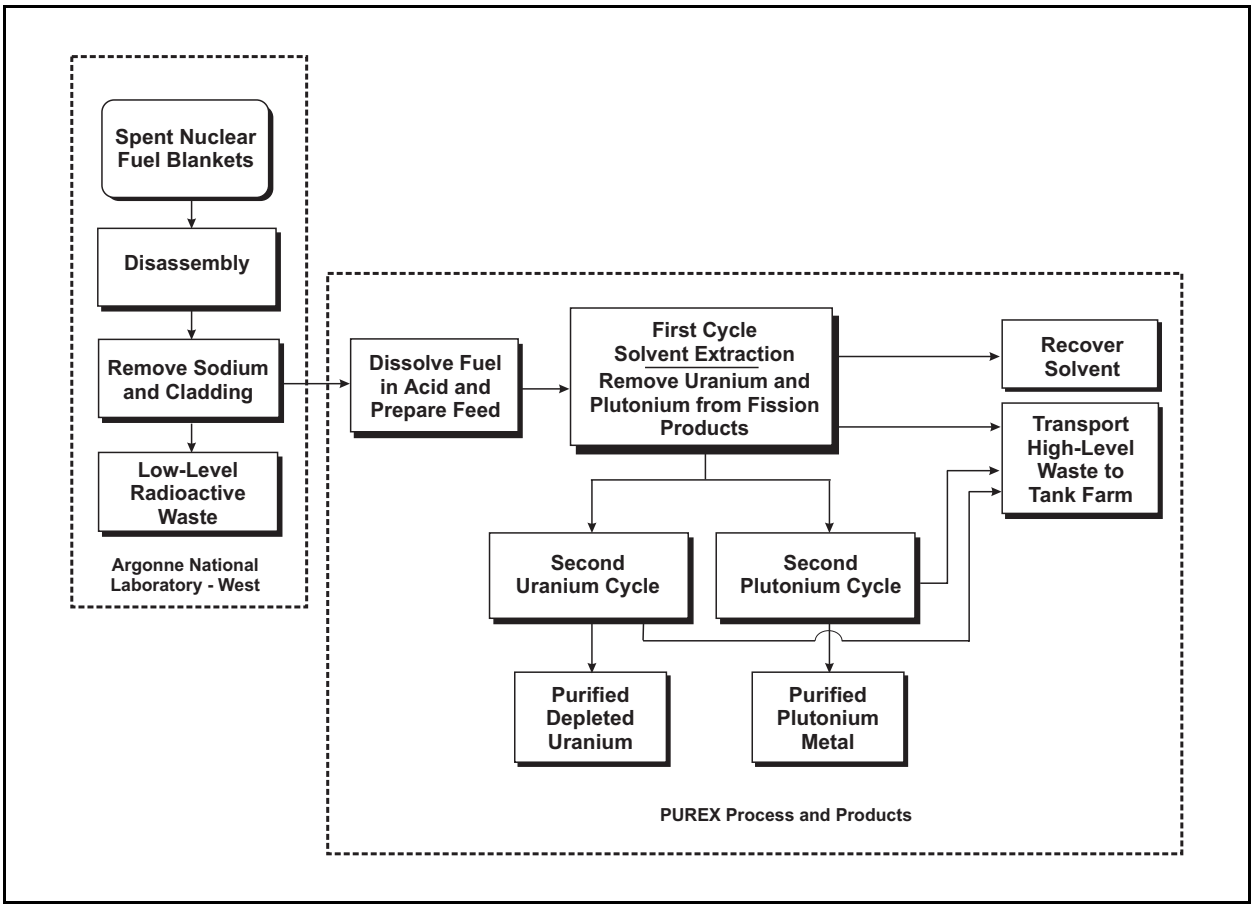

Figure C–2 PUREX Process Flow Diagram at SRS

Sodium recovered during the cleaning process would contain some fission products, most notably cesium-137. This 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 rapidly react with oxygen and water to form aqueous sodium hydroxide. Carbon dioxide gas would then 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 ceramic waste form described in Section C.1

Receiving and Storage at Savannah River Site: The packages of blanket fuel pins from ANL-W would be received and stored in Building 105-L until transfer to the F-Canyon for stabilization using the PUREX process.

PUREX Unit Operations: The EBR-II and Fermi-1 blanket fuel pins would be processed using the traditional PUREX process. This process consists of several major operations referred to as "unit operations," which yield two products, uranium and plutonium (in solution form). The unit operations are dissolution, head end, first cycle, second uranium cycle, and second plutonium cycle. Unit operations that support the product recovery process are high-activity waste, low-activity waste, and solvent recovery.

Dissolution and Head End: The irradiated material would be transferred to the canyon in casks and loaded into a large tank called a dissolver. Heated nitric acid in the tank would dissolve the blanket cores, resulting in a solution containing depleted uranium, plutonium, and fission products. Gelatin would be added to the solution, if necessary, to precipitate fuel impurities. Then the solution would be transferred to a centrifuge, where the silica and other impurities would be removed as waste. The clarified product solution from this process would be adjusted with nitric acid and water in preparation for the first cycle unit operation in the PUREX process. The waste stream generated from the process would be chemically neutralized and sent to the SRS high-level radioactive waste tanks pending further processing at the Defense Waste Processing Facility.

First-Cycle Operation: The first-cycle operation has two functions: (1) to remove fission products and other chemical impurities, and (2) to separate the solution into two product streams (i.e., uranium and plutonium) for further processing. This separation process occurs as the product solution passes through a series of equipment consisting of a centrifugal contactor and mixer-settler banks. Before the introduction of the product solution, flows of solvent and acid solution would be started through the equipment. After an equilibrium condition had been established, the product solution would be introduced. The chemical properties of the acid/solvent/product solutions in contact with each other would cause the fission products to separate from the uranium and plutonium. Later in the first cycle process, the plutonium would be separated from the uranium in a similar manner. The first cycle would produce four process streams: a plutonium-containing solution (with some residual fission products), which would be sent to the second plutonium cycle; a uranium-containing solution (with some residual fission products), which would be sent to the second uranium cycle; a solvent stream, which would be sent to a solvent recovery cycle; and an aqueous acid stream, which would contain most of the fission products and would be sent to the SRS high-level radioactive waste tanks, pending further processing at the Defense Waste Processing Facility.

Second Uranium Cycle: In the second uranium cycle, the uranium-containing solution coming from the first cycle would be purified further in a manner similar to that described for the first cycle. The purified solution would be transferred to storage tanks. Eventually, the uranium would be converted to uranium oxide and stored in 208-liter (55-gallon) drums. The uranium oxide would be stored for future use. The solution containing the residual fission products would be sent to SRS high-level radioactive waste tanks, pending further processing at the Defense Waste Processing Facility.

Second Plutonium Cycle: In the second plutonium cycle, the plutonium-containing solution coming from the first cycle would be further purified in a manner similar to that described for the first cycle. The purified solution would be converted to plutonium metal in the FB-Line, prepackaged into cans, packaged in an outer container, and placed into a vault for long-term storage, pending disposition in accordance with decisions reached in the *Surplus Plutonium Disposition Draft Environmental Impact Statement* (DOE 1998a). The solution containing the residual fission products would be sent to the SRS high-level radioactive waste tanks, pending further processing at the Defense Waste Processing Facility.

Other Unit Operations: The unit operations for high- and low-activity waste would reduce the volume of the aqueous streams containing fission products. The streams originate with primary separation process unit operations such as the first cycle. The fission products would be separated and sent to the high-level radioactive waste tanks. The volume reduction process would be accomplished using a series of evaporators in the canyons. The solvent recovery unit operation would recover and recycle the solvent that is used in the first cycle by removing impurities from the solvent. The purified solvent would be returned to the first cycle for reuse, and the impurities would be transferred to low-activity waste for processing (DOE 1994).

Technology Maturity: The PUREX process is considered to be a mature technology. It has been used throughout the world since 1954 to separate and purify uranium and plutonium from fission product-containing spent nuclear fuel and irradiated uranium targets.

C.3 HIGH-INTEGRITY CAN

The high-integrity can option is being considered for EBR-II and Fermi-1 blanket elements. The high-integrity can is made from Hastelloy Alloy C-22 metal alloy pipe having a 13.7 centimeter (5 inch) diameter and variable length and a pipe wall thickness of 0.655 centimeter (0.258 inch). The high-integrity cans are designed for dry hot cell loading with a lid adaptable to wet load and vacuum drying. The lid on each can has a threaded design to accommodate the partial loading of the spent nuclear fuel into the can at different times. The threaded lid prevents spillage of can contents during interim storage (DOE 1998b). After packaging the fuel, the cans would be placed in standardized canisters of about 46 to 61 centimeters (18 to 24 inches) in length and 61 centimeters (24 inches) in diameter for ultimate disposal in a repository (Shaber 1998). A diagram of the high-integrity can process is shown in **Figure C–3**.

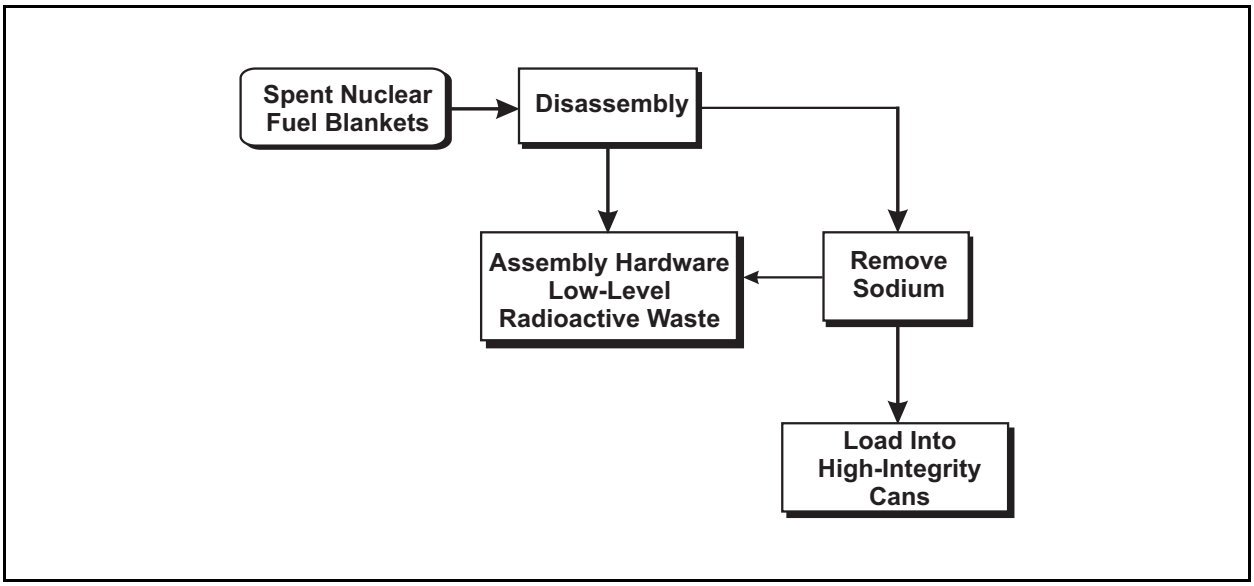

Figure C–3 High-Integrity Can Process Flow Diagram

Disassembly: Although the blanket assemblies have been mostly disassembled, there may be some assemblies that need to be removed. The assembly hardware would be separated from the blanket fuel pins by cutting the assemblies and physically separating the fuel elements. The fuel elements would be placed into a container for transfer to an argon-atmosphere hot cell for the remaining process steps. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at ANL-W. This is a normal waste stream for ANL-W operations, and the separated hardware would be handled in accordance with normal site waste management practices.

Sodium Removal: The fuel blanket elements would be brought into an argon-atmosphere hot cell where the ends of the elements would be cut off to expose the sodium within the cladding. The fuel elements would be placed into a crucible and loaded into an induction furnace. The temperature in the furnace would be raised above the melting point of sodium (200 °C [390 °F]) and the molten sodium drained into a collection tank. With most of the sodium removed, the temperature would be raised to 500 °C (930 °F) and a 10^{-4} Torr vacuum would be applied to the chamber. This would volatilize the residual sodium, allowing the sodium vapor to be drawn away from the fuel. The vapor-phase sodium would be condensed in a trap and combined with the sodium, which had been drained into the collection tank, pending further processing.

Sodium recovered during the cleaning process would contain some fission products, most notably cesium-137. This 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 rapidly react with oxygen and water to form aqueous sodium hydroxide. Carbon dioxide gas would then 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 ceramic waste form described in Section C.1.

Loading into High-Integrity Cans: The blanket elements would be packaged in a standard sized can fabricated from Hastelloy Alloy C-22, or possibly some other highly corrosion-resistant materials such as titanium Grade-12. Hastelloy Alloy C-22 (UNS N06022) is an alloy of nickel, chromium, and molybdenum that is highly corrosion resistant due to its high chromium (22 percent) and molybdenum (13 percent) content.

The high-integrity can would be placed in dry storage at an appropriate location. If transportation is required, the cans would be packaged into shipping casks. Prior to shipment to a geologic repository, the high-integrity can containing spent nuclear fuel would be placed into a standardized canister, an overpack designed to provide additional containment within the waste package under repository conditions.

Direct Disposal of Sodium-Bonded Spent Nuclear Fuel (with Minimal Treatment): Direct disposal of sodium-bonded spent nuclear fuel is currently precluded by DOE policy concerning acceptance of Resource Conservation and Recovery Act (RCRA)-designated mixed waste (which contains both hazardous and radioactive waste). In the absence of such a policy, sodium-bonded spent nuclear fuel (driver and blanket) could be cleaned of surface sodium, packaged in high-integrity cans without removal of metallic sodium from the interior of the fuel elements, and directly disposed of in a Federal repository. The high-integrity cans would be placed into a standardized canister designed to promote containment under repository conditions.

Technology Maturity: Packaging materials in a high-integrity can is considered to be a mature technology. These cans would be made from highly corrosion-resistant materials and would be designed to provide exceptional protection from external environments.

C.4 Melt and Dilute Process

The melt and dilute process is being considered for driver and blanket fuel elements. Three process options are being considered: (1) melting bare uranium blanket fuel pins with aluminum, (2) melting blanket fuel elements with cladding and additional stainless steel, and (3) developing a new melt and dilute process capable of handling the sodium volatilized form, processing chopped driver fuel elements that could not be declad and completely cleaned of sodium. Processing activities would be conducted in the Hot Fuel Examination Facility at ANL-W or in Building 105-L at SRS. A diagram of the melt and dilute process flow for the first two options is shown in **Figure C–4**. A process flow diagram for the third option is shown in **Figure C–5**.

Disassembly of Blanket Fuel Elements at ANL-W: Although the blanket assemblies have been mostly disassembled, there may be some assemblies that need to be removed. The assembly hardware would be separated from the blanket fuel elements by cutting the assemblies and physically separating the fuel elements. The fuel elements would be placed into a container for transfer to an argon-atmosphere hot cell for the remaining process steps. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at ANL-W. This is a normal waste stream for ANL-W operations, and the separated hardware would be handled in accordance with normal site waste management practices.

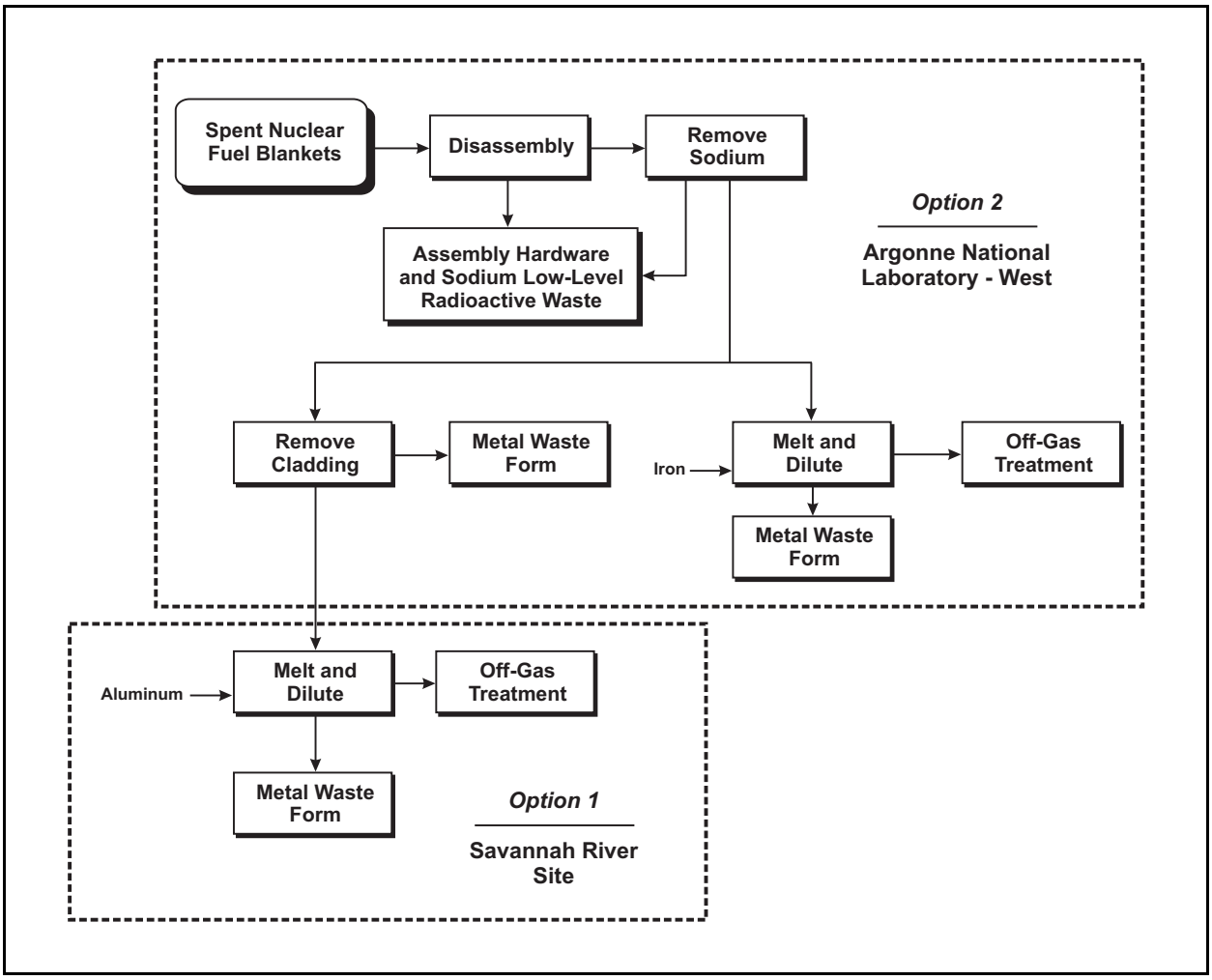

Figure C–4 Melt and Dilute Process Flow Diagram for Options 1 and 2

Sodium Removal and Processing at ANL-W: Blanket fuel elements would be brought into an argon-atmosphere hot cell where the ends of the elements would be cut off to expose the sodium within the cladding. The elements then would be cut into segments less than 61 centimeters (24 inches) in length. The fuel elements would be placed into a crucible and loaded into an induction furnace. The temperature in the furnace would be raised above the melting point of sodium (200 °C [390 °F]) and the molten sodium drained into a collection tank. With most of the sodium removed, the temperature would be raised to 500 °C (930 °F) and a 10^{-4} Torr vacuum would be applied to the chamber. This would volatilize the residual sodium, allowing the sodium vapor to be drawn away from the fuel. The vapor-phase sodium would be condensed in a trap and combined with the sodium, which had been drained into the collection tank, pending further processing.

Sodium recovered during the cleaning process would contain some fission products, most notably cesium-137. This 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 rapidly react with oxygen and water to form aqueous sodium hydroxide. Carbon dioxide gas would then 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 ceramic waste form described in Section C.1.

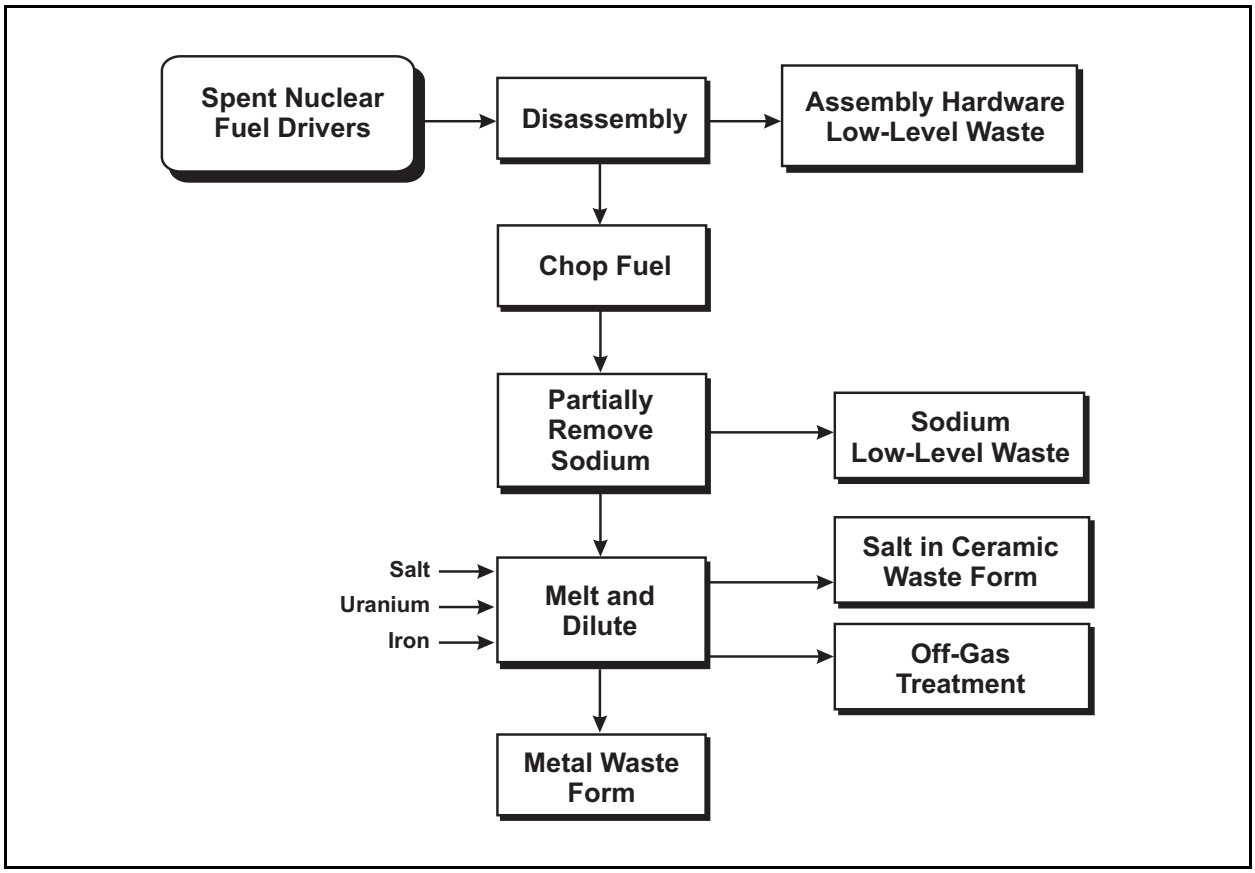

Figure C–5 Melt and Dilute Process Flow Diagram for Option 3

Decladding and Packaging Blanket Pins for Shipment to SRS: In the first melt and dilute processing option, blanket pins that would be sent to SRS would be mechanically pushed out of the stainless steel cladding after all the sodium bond had been eliminated. These blanket pins would be packed into aluminum canisters in the Hot Fuel Examination Facility. The canisters, approximately 10 centimeters (4 inches) in diameter and 61 centimeters (24 inches) in length, would be backfilled with an inert gas and sealed. Each canister would contain about 60 kilograms (130 pounds) of depleted uranium fuel pins. The canisters would be placed in a NAC-LWT cask for shipment to SRS.

Receiving and Storage at SRS: The blanket pins from ANL-W would be received at the L-Reactor Disassembly Basin in Building 105-L for storage until transfer to the processing facility.

Melt and Dilute Process at SRS: Blanket pins would be transferred to the treatment facility in Building 105-L where the material would be unloaded and validated. The fuel pieces and aluminum would be loaded into an induction furnace where they would be heated to approximately 1,000 °C (1,830 °F). This temperature is significantly in excess of the aluminum-uranium eutectic temperature to initiate the melting and have it proceed within a reasonable time. Sufficient aluminum would be added to make an aluminum-uranium alloy with a composition of about 70 percent aluminum and 30 percent uranium. The metal alloy would be cast into an ingot, sampled, and packaged into canisters. The canisters would be evacuated, filled with inert gas, sealed by welding, and transferred to storage pending disposition in a geologic repository. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. The filters would be disposed of as low-level or high-level radioactive waste, as appropriate.

Melt and Dilute Process at ANL-W: In the second melt and dilute processing option, blanket elements recovered from the sodium removal process would be placed in an induction furnace crucible with additional radioactive waste steel. Sufficient steel would be added to make an alloy with a composition of about 50 percent each of uranium and steel. The furnace would be heated to approximately 1,400° C (2,550° F) to melt the uranium, after which the steel would slowly be dissolved into the uranium pool. The mixture would be electromagnetically stirred to a uniform composition. The metal alloy would be cast into an ingot, sampled, and packaged for interim storage at the Radioactive Scrap and Waste Facility. An off-gas system would capture the volatile and semi-volatile fission products for stabilization and processing into waste forms suitable for disposal. The filters would be disposed of as low-level or high-level radioactive waste, as appropriate.

Melt and Dilute Process for Driver Fuel at ANL-W: The third melt and dilute option would be for sodium-bonded driver fuel. For driver fuel with burnup of 1 - 3 percent or higher, the coolant gap between the uranium and cladding is completely closed by fuel swelling and material interdiffusion between the cladding and the fuel pin. Clean separation of the cladding and uranium is no longer possible once the interdiffusion process has begun. During the swelling process, the fuel becomes very porous. At discharge from the reactor, 15 to 20 percent of the fuel's porosity contains trapped sodium with dissolved fission products. Because the cladding and sodium cannot be readily removed from the driver fuel elements, a melt and dilute process capable of handling the volatilized sodium would be required to process driver fuel elements. Sodium-based uranium oxide, uranium carbide, and uranium nitride fuels cannot be treated using the melt and dilute process because of their high melting points.

Some of the sodium in driver fuel elements would be removed in a manner similar to the way sodium would be removed from blanket fuel elements, i.e., the fuel would be cut into smaller pieces to allow some sodium to drain away and then the fuel pieces would be heated under vacuum to volatilize additional sodium. However, all of the sodium in driver fuel elements could not be removed by these processes.

The cut driver fuel elements, which would still retain some sodium, would next be covered with a layer of low-melting-temperature salt containing uranium chloride to oxidize the molten sodium. Depleted uranium would be added in the ratio of about 2.5:1 to reduce the enrichment to less than 20 percent uranium-235. Radioactive waste steel would be added in equal weight to the uranium to complete the mix. The furnace would then be heated to a temperature of 1,000 to 1,400 °C (1,800 to 2,550 °F). The molten salt would capture sodium vapors escaping from the fuel elements as they melt, protecting the downstream components from the sodium. After volatilization of the sodium and reaction with the molten salt, a vacuum would be applied to the furnace to volatilize the salt, which would be condensed and partially reused. The salt would be stabilized in the ceramic form described in Section C.1. The molten metal would be stirred to achieve a uniform composition and cast into an ingot, placed into a container and stored. An off-gas system would capture the volatile and semi-volatile fission products for stabilization and processing into waste forms suitable for disposal. The filters would be disposed of as low-level or high-level radioactive waste, as appropriate.

Technical Maturity: The melt and dilute process was developed for treating aluminum-based spent nuclear fuel at SRS and is DOE's preferred technology for treating that type of spent nuclear fuel. The melt and dilute process for stainless steel-clad spent nuclear fuel would require operating temperatures of approximately 1,400 °C (2,550 °F), compared with about 1,000 °C (1,830 °F) for aluminum-based spent nuclear fuel. Induction-heated melters that can achieve the higher temperatures required for stainless steel have been demonstrated at ANL-W. Technology development would be required to demonstrate capturing the quantities of sodium present in the driver fuel assemblies in a molten salt.

C.5 DIRECT PLASMA ARC TREATMENT PROCESS

The plasma arc treatment technology (DOE 1998c) would use a plasma torch to melt and oxidize the spent nuclear fuel in conjunction with depleted uranium oxide and other ceramic-forming materials, as necessary.

The fuel would be fed into the process with minimal sizing or pretreatment. The plasma arc would cut the fuel assemblies into small pieces and heat the fuel to temperatures at least as high as 1,600 °C (2,900 °F) to melt and oxidize it in a rotating furnace. Ceramic material would be added, as necessary, with the mixture homogenized by the torch. When melting and oxidation were complete, the rotating furnace would slow and the melt would fall into molds prepared to receive it. A diagram of the plasma arc treatment process flow is shown in **Figure C–6**.

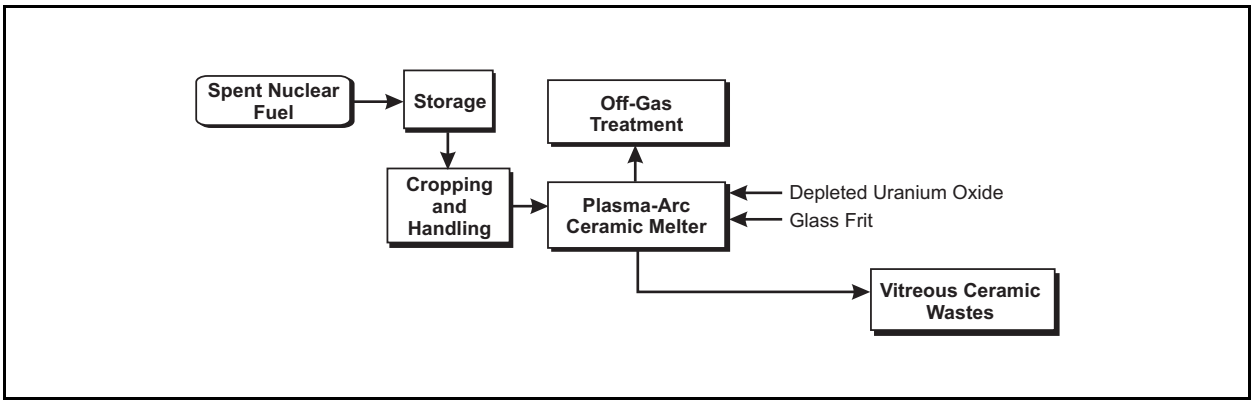

Figure C–6 Plasma Arc Treatment Process Flow Diagram

Metallic fuels such as EBR-II 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. Criticality issues would be addressed by limiting the process to batch runs of preselected quantities of fissile material, by the addition of the depleted uranium, and by the addition of 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, with the filter and treatment media recycled into the plasma arc furnace for incorporation into the ceramic product.

Technology Maturity: The plasma arc process is a developmental technology that has not been demonstrated for stabilization of spent nuclear fuel.

C.6 GLASS MATERIAL OXIDATION AND DISSOLUTION SYSTEM

Glass Material Oxidation and Dissolution System (GMODS) uses lead oxide to convert unprocessed spent nuclear fuel directly to borosilicate glass using a batch process. A diagram of the GMODS process flow is shown in **Figure C–7.**

Metal Oxidation: The principal piece of equipment for GMODS would be an induction-heated, cold-wall melter, which is commercially used to convert corrosive or high-melting metals to ultrapure materials. The melter, operating at 1,000 to 1,200 °C (1,830 to 2,200 °F), would be charged with a molten glass consisting of lead oxide and boron oxide. Oxides and amorphous components of the spent nuclear fuel would directly dissolve into the glass. Metals, which do not normally dissolve in glass, would be converted to oxides by the lead oxide. Boron oxide, a neutron poison, is a common agent for dissolving oxides into glass. Criticality concerns would be addressed by diluting the uranium-235 enrichment with depleted uranium and using boron oxide as a dissolving agent (DOE 1998c).

On feeding the spent nuclear fuel into the melter, the uranium, plutonium, and other metals would be oxidized and dissolved in the molten glass. The oxidation of the metals would convert the lead oxide to metallic lead,

which would sink to the bottom of the melter. Radioactive off-gases produced during this process would be filtered. The filters would be managed as high-level, low-level, or mixed waste, as appropriate.

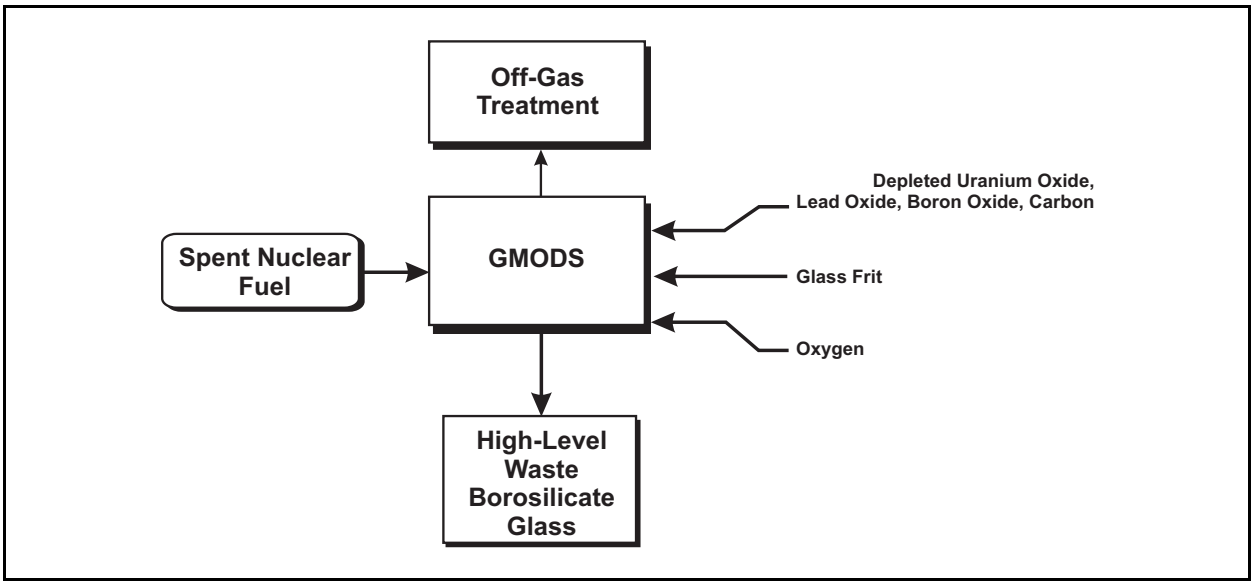

Figure C–7 Glass Material Oxidation and Dissolution (GMODS) Process Flow Diagram

Conversion of Lead to Lead Oxide: After decanting the glass, the melter would be recharged with boron oxide and, if necessary, lead oxide. Oxygen would be piped into the system to convert the metallic lead at the bottom of the melter back to lead oxide. Therefore, lead would be an oxygen carrier that would not leave the system.

Glass Waste Form: The resulting glass mixture would not have qualities necessary for long-term durability, so silicon oxide (glass frit) would need to be added to increase the durability of the high-level radioactive waste borosilicate glass. The silicon oxide would not be part of the initial melter charge because its properties are not conducive to rapid oxidation-dissolution of spent nuclear fuel. Unreduced lead oxide could limit the durability of the glass, and increase volume, so carbon would be added to the melt to reduce the excess lead oxide (DOE 1998c).

Technology Maturity: GMODS process has been developed by DOE for stabilization of radioactive wastes. At this time, it has only been tested in small-scale laboratory experiments.

C.7 CHLORIDE VOLATILITY PROCESS

Chloride Volatility (NAS 1998) is an advanced treatment technology that was investigated at the Idaho National Engineering and Environmental Laboratory (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. A diagram of the chloride volatility process flow is shown in **Figure C–8**.

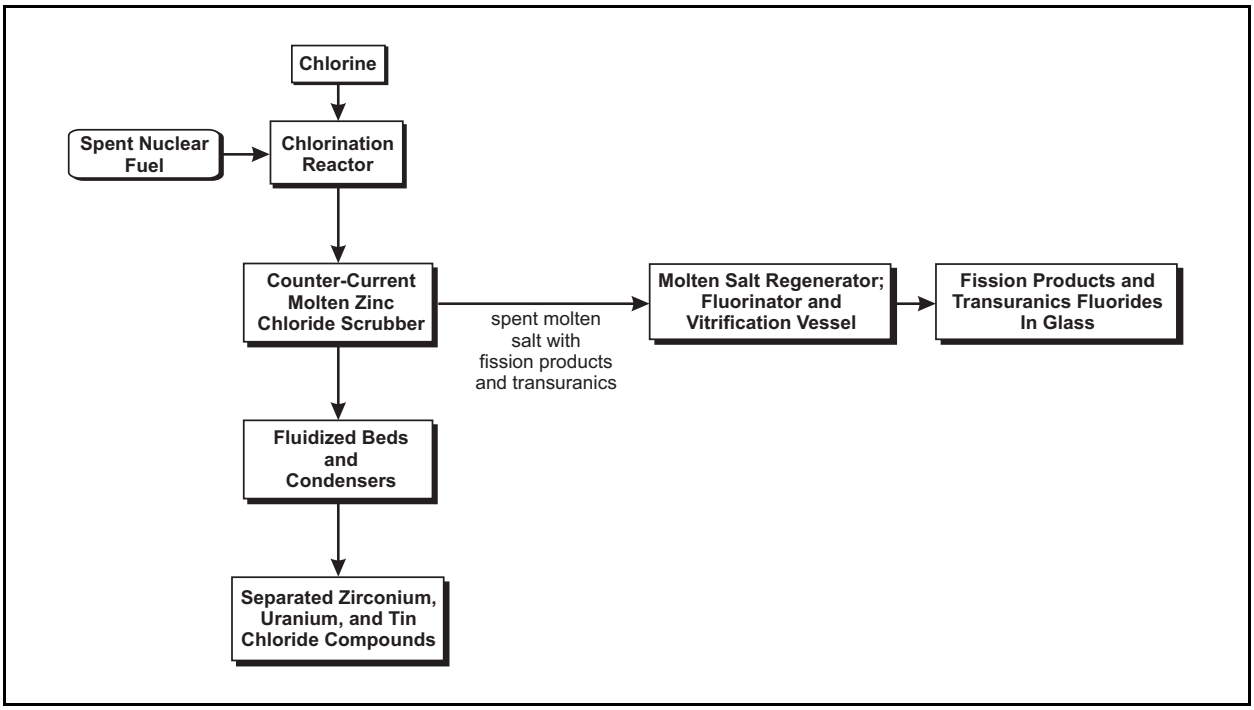

Figure C–8 Chloride Volatility Process Flow Diagram

The chloride volatility process would consist of four operations:

(1) A high-temperature chlorination step that would operate at approximately 1,500 °C (2,730 °F) and would convert fuel and cladding materials to gaseous chloride compounds

(2) A molten zinc chloride bed that would remove the transuranic chlorides and most of the fission products and would operate at approximately 400 °C (750 °F)

(3) A series of fluidized beds and condensers that would operate at successively lower temperatures to condense zirconium tetrachloride, uranium hexachloride, and stannous tetrachloride

(4) A zinc chloride regeneration/recycle process

The transuranic and fission product chlorides would then be converted to either fluorides or oxides for final disposal. Argon carrier gas and unreacted chlorine gas would be recycled, the chlorine content adjusted, and the stream split and passed through the unit operations in a continuous closed loop. Periodic shutdowns of the coupled unit operations would occur for batch removal of fission product xenon and krypton gases from the carrier gas (such as by cryogenic distillation), batch transfer of the molten salt to the molten salt regenerator, and batch removal of nonradioactive constituents and uranium from the condensers.

The small quantity of fission-product/transuranic-product high-level radioactive waste would be converted into a waste form for repository disposal. The conversion steps to a glass or glass-ceramic form could involve fluorination and melting with glass frit additives, or conversion to oxides by heating at 1,000 °C (1,830 °F) with boric acid.

In the chlorination step, the rate of reaction would be controlled by the feed rate of chlorine, and the temperature would be controlled by appropriate blending of argon gas with chlorine. An oxygen scavenger, such as carbon monoxide, would be added as needed to prevent formation of oxychlorides when oxides are

present. A carbon dioxide absorption bed in the off-gas system would collect the carbon dioxide that would be formed. Zinc chloride would be used for the scrubber medium because its low melting point and favorable vapor pressure would permit its use to scrub the chlorinator off-gas at a low temperature, while its volatility at 725 °C (1,337 °F) would allow evaporative separation from the radioactive waste chlorides for subsequent recycle.

Theoretical chloride volatilities have been used to postulate the equipment sizing and operating parameters. Because of the lack of any experimental basis, significant concerns exist about the distribution of chloride compounds for multivalent elements such as uranium and plutonium. These concerns, in turn, lead to potential uncertainties in separation capabilities and overall flowsheet performance. The use of halides, either fluorides or chlorides, for the transuranic and fission product elements raises questions about the use of a glass or vitrified waste form. A proposal to use boric acid at 1,000 °C (1,830 °F) allays some of those concerns (LITC 1996).

Technology Maturity: The chloride volatility process has not progressed beyond the conceptual design stage. No laboratory experiments have been conducted.

C.8 REFERENCES

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APPENDIX D
SODIUM-BONDED FUEL CHARACTERISTICS

D.1 BACKGROUND

D.1.1 General Characteristics

The sodium-bonded spent nuclear fuel addressed in this environmental impact statement (EIS) is primarily from the operation of the Experimental Breeder Reactor-II (EBR-II) and Fermi-1 breeder reactors (a small percentage of the spent nuclear fuel is derived from other sources). This fuel can be categorized as either driver fuel or blanket fuel. Driver fuel consists of highly enriched uranium alloy (alloy of uranium in zirconium or fissium[1]) fuel. (Natural uranium consists of mostly uranium-238, containing approximately 0.7 weight percent uranium-235; low-enriched uranium contains less than 20 weight percent uranium-235; highly enriched uranium contains greater than or equal to 20 weight percent uranium-235.) As a fissile material, uranium-235 is capable of undergoing fission (splitting into two major fragments called fission products) releasing energy and additional neutrons when struck by a neutron. This enriched uranium core produces the majority of the neutrons that power (drive) the reactor and breeding in the blanket, hence the name driver fuel. In the blanket region, uranium-238 from either natural uranium or depleted uranium, which has less than 0.3 weight percent uranium-235, capture neutrons to produce fissile materials, such as plutonium-239. In this manner, breeder reactors can produce (or breed) more fissile material than they consume.

The uranium in nuclear fuels is clad with a metal to protect it from chemical reactions with the coolant and to prevent the release of fission products to the coolant. Zirconium, stainless steel, and aluminum are common cladding materials. Most of the spent nuclear fuel analyzed in this EIS is clad with stainless steel.

Inside the cladding, the fuel is often in the form of a ceramic, an alloy that combines uranium with other metals such as zirconium, metallic uranium, or an oxide, carbide, silicide, nitride, or other form. The fuel can be fabricated as parallel plates, concentric tubes, bundles of rods or pins, or other designs. Each individual fuel item is referred to as a fuel element. Multiple fuel elements are typically combined into an assembly or subassembly. Each assembly/subassembly has mounting and lifting hardware, structures to direct coolant, and in some cases the capability to install neutron absorbing material and instrumentation. Most of the fuel elements addressed by this EIS are uranium alloy rods or pins. In order to improve the transfer of heat from the uranium matrix where the heat is generated to the cladding, the gap between the fuel and the cladding has been filled with a small amount of metallic sodium.

Usually a number of fuel assemblies/subassemblies make up a reactor core. Blanket assemblies/subassemblies placed around the reactor driver core for breeding or shielding are similar in design to driver fuel. An axial blanket may be placed above and below the reactor core and a radial blanket may be placed at the perimeter of the reactor core.

[1] *Fissium is a mixture of noble metals (molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium).*

D.1.2 Recent Spent Nuclear Fuel Management Actions

In 1992, DOE decided to phase out defense-related spent nuclear fuel reprocessing. Subsequently, the Department began to establish programs to manage DOE spent nuclear fuel that were no longer based on the production of strategic nuclear material. DOE identified the initial components of this plan in the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995) (hereafter referred to as the Programmatic Spent Nuclear Fuel EIS). The Record of Decision for this EIS (60 FR 28680) stated, in part, that DOE would consolidate the management of its aluminum-clad spent nuclear fuel at the Savannah River Site (SRS), leave the Hanford production spent nuclear fuel at Hanford, and would consolidate nonaluminum-clad fuels at the Idaho National Engineering and Environmental Laboratory (INEEL). This Record of Decision was amended in March 1996 (61 FR 9441). The amended Record of Decision leaves all Fort St. Vrain spent nuclear fuel at the storage site in Colorado, all but sodium-bonded spent nuclear fuel at Hanford, and places restrictions on shipment schedules.

However, in the Programmatic Spent Nuclear Fuel EIS Record of Decision, DOE made no decisions on the technologies it would apply to the management of spent nuclear fuel at the designated storage sites. The Record of Decision stated that the selection of spent nuclear fuel stabilization technologies and the preparation of spent nuclear fuel for ultimate disposition would be the subject of site-specific and fuel-type-specific evaluations prepared in accordance with the National Environmental Policy Act (NEPA) and tiered from the Programmatic Spent Nuclear Fuel EIS (DOE 1995).

D.2 INVENTORY OVERVIEW

This EIS addresses a variety of spent nuclear fuel types that have one common characteristic, the presence of metallic sodium (or sodium and potassium). 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 spent nuclear fuel that contains metallic sodium. This EIS addresses a range of technologies that may be used to treat and manage this spent nuclear fuel for disposal. Based on composition, there are five broad categories of spent nuclear fuel to be considered: EBR-II driver spent nuclear fuel, EBR-II blanket, Fermi-1 blanket, Fast Flux Test Facility fuel, and miscellaneous spent nuclear fuel. Within each of these broad categories, there are variations within each category, but the categories can generally be described as follows:

EBR-II driver – This spent nuclear fuel is stainless steel clad highly enriched uranium in a uranium alloy, typically either fissium or zirconium. There are some variations in the specific cladding alloys, the enrichments, fuel compound alloy, dimensions, and burnup within this category. Also, there are small amounts of fuel experiments that use a different uranium compound, for example uranium carbide. This fuel type was added to the miscellaneous group.

EBR-II blanket – This spent nuclear fuel consists of stainless steel clad depleted uranium in a uranium metal form. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary difference between these blankets is dimension and burnup.

Fermi-1 blanket – This 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 difference between these blankets is dimension, elements per subassembly, and burnup. Fermi-1 blankets are similar to EBR-II blankets in enrichment

and burnup, but differ in dimension (Fermi-1 elements are larger) and form (uranium metal versus uranium-molybdenum alloy).

Fast Flux Test Facility – This group of fuel includes both irradiated and fresh driver fuels. The fuel is either uranium zirconium or plutonium/uranium zirconium, with some containing plutonium/uranium carbide and nitride. These fuels are stainless steel clad with various levels of enrichments.

Miscellaneous – This group includes experimental spent nuclear fuel from experiments irradiated in the Engineering Test Reactor and the Annular Core Research Reactor at Sandia National Laboratories. There are small quantities of experimental fuels that have metallic sodium or potassium. These fuels are highly diverse and differ in cladding, uranium compound, enrichment, and burnup.

Table D–1 provides a summary of all DOE sodium-bonded spent nuclear fuel. It should be noted that the inventories reported in Table D–1 include 0.4 metric tons of heavy metal of EBR-II driver fuel and the 1.2 metric tons of EBR-II blanket fuel that are being treated as part of the demonstration program.

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

Fuel Type	*Storage Volume (cubic meters)*[a]	*Total EOL Fissile (kilograms)*	*EOL Metric Tons of Heavy Metal*
EBR-II Driver	58	2,030	3
EBR-II Blanket	13	285	22
Fermi-1 Blanket	19	130	34
Fast Flux Test Facility	8	175	0.3
Miscellaneous	3	60	0.1
Total	101	2,680	60

EOL = End of Life
[a] Storage volume (ANL 1999).

By any measure, the majority of the spent nuclear fuel consists of EBR-II driver, EBR-II blanket, and Fermi-1 blanket fuel. **Table D–2** provides a summary of the fraction of spent nuclear fuel in each category by a variety of different measures. As shown, the percentages vary considerably depending upon the measure used for comparison.

Table D–2 Comparison of Sodium-Bonded Spent Nuclear Fuel[1] by Different Measures

	Volume (cubic meters)	*Total EOL Fissile (kilograms)*	*EOL metric tons of heavy metal*
EBR-II Driver	58 percent	75 percent	5 percent
EBR-II Blanket	13 percent	11 percent	37 percent
Fermi-1 Blanket	19 percent	5 percent	57 percent
Fast Flux Test Facility	9 percent	7 percent	0.5 percent
Miscellaneous	3 percent	2 percent	< 0.1 percent
Total	100 percent	100 percent	100 percent

Note: Values may not add to exactly 100 percent due to rounding.

The radionuclide inventory of the spent nuclear fuel varies widely due to differences in the construction, function and operational history of the spent nuclear fuel. Therefore, radionuclide inventory estimates were developed for EBR-II driver fuel (including a separate estimate for the experimental driver fuel), EBR-II blanket, Fermi blanket, and Fast Flux Test Facility experimental fuel (SAIC 1999). **Table D–3** provides a summary of plutonium and sodium content for each fuel type. **Table D–4** provides a summary of the radionuclides for each of the fuel types.

Table D–3 Plutonium and Sodium Content in Sodium-bonded Fuel

Spent Nuclear Fuel Type	*Plutonium (kilograms)*	*Sodium (kilograms)*
EBR-II Driver	19	83
EBR-II Blanket	250	176
Fermi-1 Blanket	7	365
Fast Flux Test Facility	3	7
Miscellaneous	Not Available	31
Total	279	662

Table D–4 Principal Radionuclide Activities per Kilogram of Heavy Metal [a]

Elements	*Isotope*	*EBR-II Driver [b]*	*EBR-II Radial Blanket [c]*	*EBR-II Exp. Driver Fuel*	*Fermi-1 Blanket*	*FFTF Driver*
Tritium	H3	1.22e+00	7.12e-03	1.16e+00	7.56e-05	1.90e+00
Carbon	C14	1.99e-04	5.97e-05	9.54e-04	1.05e-08	6.74e-04
Iron	Fe55	4.87e+0	9.01e-02	5.11e+00	2.69e-05	9.89e+00
Nickel	Ni63	2.29e-01	3.06e-03	1.52e-01	4.82e-05	4.91e-02
Krypton	Kr85	1.89e+01	5.20e-02	1.65e+01	6.63e-04	2.39e+01
Strontium	Sr90	1.97e+02	8.07e-01	1.71e+02	1.63e-02	2.41e+02
Yttrium	Y90	1.97e+02	8.07e-01	1.71e+02	1.63e-02	2.41e+02
Ruthenium	Ru106	1.51e+00	1.35e-01	2.67e+00	7.02e-10	3.95e+00
Rhodium	Rh106	1.51e+00	1.35e-01	2.67e+00	7.02e-10	3.95e+00
Cadmium	Cd113M	4.64e-02	7.12e-04	5.11e-02	2.86e-06	6.59e-02
Antimony	Sb125	2.96e+00	2.31e-02	2.98e+00	2.92e-06	4.72e+00
Tellurium	Te125M	1.23e+00	9.51e-03	1.23e+00	1.20e-06	1.89e+00
Iodine	I129	7.35e-05	1.44e-06	6.85e-05	1.26e-08	8.98e-05
Cesium	Cs134	1.76e+00	1.34e-02	1.93e+00	6.66e-09	4.19e+00
	Cs137	2.21e+02	1.73e+00	1.99e+02	2.43e-02	2.72e+02
Barium	Ba137M	2.09e+02	1.64e+00	1.88e+02	2.30e-02	2.57e+02
Cerium	Ce144	2.96e+00	6.27e-02	5.55e+00	6.60e-12	9.88e+00
Praseodymium	Pr144	2.96e+00	6.27e-02	5.55e+00	6.60e-12	9.88e+00
Promethium	Pm147	8.26e+01	4.07e-01	8.02e+01	8.10e-05	1.28e+02
Samarium	Sm151	5.34e+00	1.00e-01	5.00e+00	1.31e-03	6.49e+00
Europium	Eu154	5.67e-01	7.34e-03	6.28e-01	7.70e-07	9.69e-01
	Eu155	3.81e+00	4.81e-02	3.97e+00	6.71e-05	5.28e+00
Thorium	Th228	5.14e-05	1.55e-07	5.61e-05	1.32e-10	7.39e-05
Uranium	U234	4.04e-02	1.33e-06	3.71e-02	3.20e-08	4.07e-02
	U235	1.31e-03	3.77e-06	1.20e-03	7.48e-06	1.23e-03
	U236	1.21e-03	4.24e-06	1.04e-03	1.09e-07	1.41e-03
	U238	1.11e-04	3.27e-04	1.20e-04	3.31e-04	1.17e-04
Neptunium	Np237	2.89e-04	8.37e-06	2.87e-04	2.28e-07	4.01e-04
Plutonium	Pu238	1.66e-01	9.39e-03	2.33e-01	3.34e-06	3.04e-01

Elements	*Isotope*	*EBR-II Driver* [b]	*EBR-II Radial Blanket* [c]	*EBR-II Exp. Driver Fuel*	*Fermi-1 Blanket*	*FFTF Driver*
	Pu239	2.69e-01	7.53e-01	1.61e+00	1.34e-02	7.39e-01
	Pu240	9.11e-03	5.18e-02	7.54e-01	1.12e-05	1.23e-01
	Pu241	2.22e-02	2.10e-01	1.44e+01	3.54e-07	1.60e+00
Americium	Am241	3.91e-04	1.63e-02	3.59e-01	3.46e-08	5.16e-02
Americium	Am242M	3.313e-07	1.69e-04	2.18e-03	7.84e-14	1.40e-04
Total	**Ci/kg**[d]	**9.85e+02**	**7.18e+00**	**8.77e+02**	**9.59e-02**	**1.24e+03**
Total Heavy Metal Mass	metric ton	3.1	22.4	0.2	34.2	0.25

[a] Activities are in curies per kilogram of heavy metal, as of January 1, 2000.
[b] Inventory of Mark III driver fuel is bounding fuel for all EBR-II driver fuel types.
[c] Representative for all EBR-II Blanket fuel.
[d] Curie per kilogram of heavy metal.
[e] 1.13e+03 = 1.13 x 10^3, or 1,130.

The following sections provide a more detailed description of each category of spent nuclear fuel.

D.3 EBR-II SPENT NUCLEAR FUEL

D.3.1 Reactor Background

EBR-II was a research and test reactor used to demonstrate the engineering feasibility of a sodium-cooled, Liquid Metal Fast Breeder 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 its operation, numerous fuel designs were tested in EBR-II. The reactor operating power level was 62.5 megawatts-thermal.

D.3.2 Description of EBR-II Spent Nuclear Fuel

The EBR-II reactor consisted of an enriched driver core surround by depleted blanket assemblies. The reactor originally had an upper and lower axial blanket above and below the driver core, as well as a radial blanket around the perimeter of the driver core. It later operated with a radial blanket only. In addition, various experimental assemblies where placed into the core for testing. The following sections describe the driver fuel (including experiments) and blanket assemblies.

D.3.2.1 Driver Fuel

Standard Driver Fuel

The driver fuel contains highly enriched uranium (enrichment of up to 78 weight percent). When the fuel is 'spent,' the enrichment (ratio of uranium-235 to total uranium) ranges between 55 percent and 76 percent.

Each driver fuel element has a metal rod (also called a fuel pin) about 36 centimeters (14 inches) long and less than 0.5 centimeters (0.2 inches) in diameter. A typical EBR-II driver fuel pin is a metal alloy of 90 percent uranium and 10 percent zirconium. This fuel pin and a small amount of metallic sodium were loaded into a 73.7-centimeter (29-inch) long stainless-steel tube (cladding) and welded shut, as shown in **Figure D–1**. This unit of fuel is called an "element." Sixty-one (in some fuels, ninety-one) fuel elements were put together in a stainless-steel hexagonal "can" to make a fuel assembly approximately 2.3 meters (92 inches) long and 5.8 centimeters (2.3 inches) across. A typical fresh (unirradiated) driver fuel assembly contains 4.5 kilograms (9.9 pounds) of uranium and a typical irradiated fuel assembly contains 4.1 kilograms (9.0 pounds).

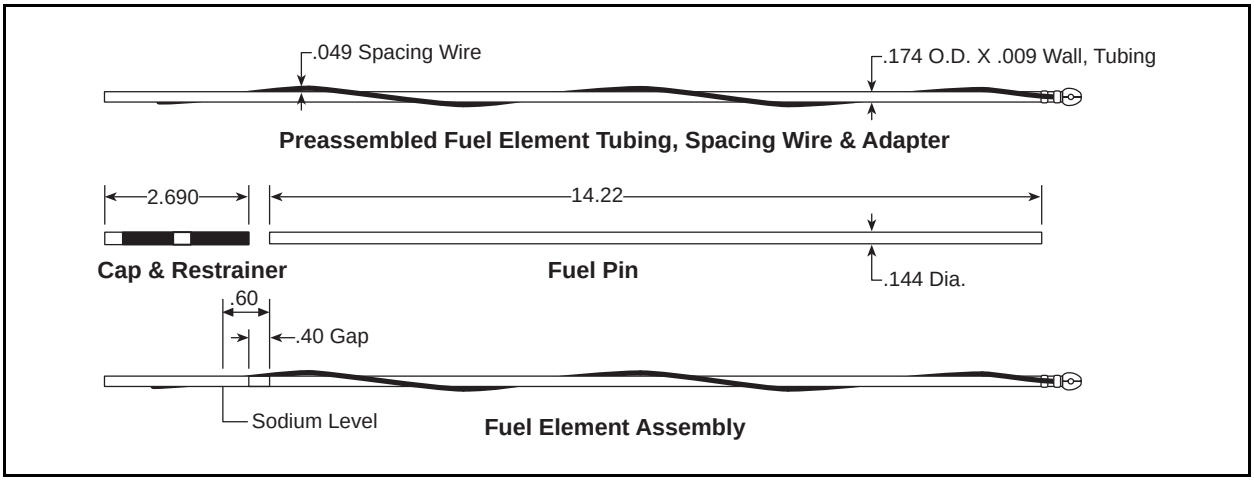


Figure D–1 Typical EBR-II Driver Element

The sodium inside driver fuel and blanket elements improves the heat transfer from the fuel to the reactor coolant through stainless steel cladding. When the driver fuel is irradiated in the reactor for some period of time, the metallic pin swells until it reaches the cladding wall. Pores form throughout the fuel pin 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 pin. 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 pin surface, including those containing sodium. Between 20 and 40 percent of the available sodium flows into the fuel pores and is inseparable from the uranium except by dissolving or melting the fuel. Further, during reactor operations, cesium-137 (an abundant radioactive fission product) dissolves in the sodium. Cesium, a reactive metal with chemical properties similar to sodium, remains with the sodium until the spent nuclear fuel is treated.

There have been numerous different fuel assemblies used in the EBR-II reactor, including a variety of experimental fuels. The types of standard spent nuclear fuel include Mark-I/IA, Mark-II/IIA, Mark-IIC/IICS, and Mark-III/IIIA. These different fuels are quite similar, but differ in terms of dimensions, enrichment, fuel alloy, and cladding material. **Table D–5** shows the range of properties for EBR-II fuel, experimental fuels, and blanket elements.

Argonne National Laboratory (ANL) has performed radionuclide projections individually for all of its spent nuclear fuel elements with the ORIGEN-RA depletion code and created a database containing inventory projections for all sodium-bonded spent nuclear fuel at ANL-W (Liaw 1998). The radionuclide inventory for a typical standard driver and experimental driver fuel element is presented in Table D–4. The driver fuel inventory is based on an average of the Mark-III elements, which are expected to have the highest inventory of the driver fuels. The EBR-II experimental driver inventory is based on the average of the experimental fuel elements that have not been processed. There may be individual elements with inventories that exceed this basis, but these inventories are well above the average for all driver assemblies.

Table D–5 Description of Unirradiated Typical EBR-II Driver and Blanket Fuel Elements

Property	*Standard Driver Fuel*	*Experimental Driver Fuel*	*Axial Blanket*	*Radial Blanket*
Element Description:				
Cladding material	SS[a]-304L, D-9, SS-316, HT9	SS-316, HT9, D-9	SS-304	SS-304
Clad outside diameter (inches)	0.179 – 0.23	0.17 - 0.29	0.38	0.49
Clad thickness (inches)	0.009 – 0.015	0.012 - 0.022	0.022	0.018
Element length (inches)	18 – 30	24 - 30	22	62
Fuel pins or rods per assembly	61 – 91	61	19	19
General Composition:				
Uranium alloy composition	U–5F[b] U–10Zr	U–10Pu–10Zr Pu/U-Carbide	uranium metal	uranium metal
Uranium-235 enrichment (percent)	67-78	up to 93	0.2	0.2
Burnup (atom percent)	up to 10	up to 18	0.014	0.2
Sodium (g/element)	1.0 – 2.0	1.0 – 2.0	~ 3	~ 20

[a] Stainless steel.
[b] An alloy of 95 weight percent uranium and 5 weight percent fissium. Fissium consists of molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium.

Experimental Fuel

EBR-II has irradiated various different experimental driver fuels in support of its own and other liquid metal fast breeder reactor fuel development programs. Over 3,000 of these fuel elements still exist. Some of these experiments investigated the use of different fuel compositions including Uranium-10Plutonium-10Zirconium, Plutonium-Carbide, Uranium-Carbide and Uranium-Oxide. Table D–5 provides the range of data applicable for experiments. While the quantity of experimental spent nuclear fuel is relatively small, it is significant because of the associated potential unique requirements. Before this fuel can be treated, the carbide and oxide forms of the fuel may have to be reprocessed and converted to metallic forms.

D.3.2.2 Axial and Radial Blanket

The blanket assemblies were made from depleted uranium, a type of uranium in which most of the fissile uranium-235 has been removed, leaving 99.7 percent uranium-238. This type of uranium will fission, but not readily, and cannot be used alone to power a nuclear reactor. Early in EBR-II's history, the blanket assemblies surrounded or "blanketed" the reactor core to demonstrate the breeding of plutonium-239, another fissile material. However, in 1967 the breeding experiment was completed and the job of reconfiguring the reactor for its role as an irradiation test facility began. By 1972, the final blanket assemblies had been moved well away from the core and replaced by a thick ring of stainless-steel reflector assemblies. In this configuration, the blanket assemblies provided shielding to protect structural materials from radiation emanating from the core.

Blanket assemblies are similar to the 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, with 3 to 5 pins placed end-to-end to make a sodium-bonded blanket element 140 centimeters (55 inches) long. Since the blanket pins are a larger diameter and longer length, 19 blanket elements comprise a blanket assembly containing approximately 47 kilograms (103 pounds) of uranium. On average, about 99 percent of the uranium remains in the spent blanket assemblies with the remaining 1 percent having been converted to fission products

and transuranic elements. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Table D–4.

Some of the EBR-II blanket assemblies have been in the reactor since it began operation more than 30 years ago. With the shutdown of EBR-II, these assemblies were unloaded from the reactor. In preparation for interim storage in the Radioactive Scrap and Waste Facility, they were cleaned to remove the few grams of sodium coolant that had adhered to the external surface as they are pulled out of the reactor.

D.3.2.3 Storage

Most of the fuel from the last seven years of EBR-II operation is presently stored in three different facilities at ANL-W: the Fuel Conditioning Facility, Hot Fuel Examination Facility, and Radioactive Scrap and Waste Facility. Previously, the spent nuclear fuel was shipped to the Idaho Nuclear Technology and Engineering Center (INTEC) (formerly Idaho Chemical Processing Plant) 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 PUREX system, went into full use at EBR-II. Approximately 6 metric tons (6.6 tons) of EBR-II fuel were processed at INTEC. When DOE stopped processing at INTEC in 1992, elements from some 500 EBR-II spent driver fuel assemblies of earlier design were left in storage pools located at INTEC. The spent nuclear fuel generated after shipments to INTEC ceased was stored at ANL-W in several facilities (Fuel Conditioning Facility, Hot Fuel Examination Facility, and Radioactive Scrap and Waste Facility).

D.4 FERMI-1 BLANKET

D.4.1 Reactor Background

The Enrico Fermi Atomic Power Plant[2] was designed and built at Monroe Beach, Michigan (30 miles southwest of Detroit) to demonstrate the feasibility of the fast breeder reactor for electric power production. Fermi-1 was a sodium cooled, fast reactor. Information was provided by Argonne National Laboratory, based upon EBR-I and EBR-II, to assist in the design of the Fermi-1 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.

On October 5, 1966, Fermi-1 experienced a coolant blockage caused by a detached piece of zirconium liner. As a result, melting occurred in 2 subassemblies and the reactor was shutdown for 3 years and 9 months. On July 18, 1970, the second Fermi-1 reactor core achieved criticality. New fuel and some of the original fuel was used for the second core. Termination of reactor operations in 1972 was not due to mechanical or technical problems, but rather due to lack of adequate financial support.

D.4.2 Blanket Description

The reactor had 2 different blanket designs: axial blanket assemblies above and below the core, and radial blanket assemblies surrounding the core. The core subassemblies (25.69 percent enriched fuel) were not bonded with sodium and are not part of the scope of this EIS. All blanket assemblies contained depleted uranium and contain a sodium bond between the uranium and the cladding. **Figure D–2** shows the radial blanket assembly. The inner and outer radial blanket assemblies had the same design and only differed in their placement in the reactor. The axial assemblies are similar, except they are shorter and have fewer, larger diameter pins. **Table D–6** provides data on both the axial and radial subassemblies.

[2]*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 is also known as Fermi, Fermi-1 or Enrico Fermi-1 (EF-1).*

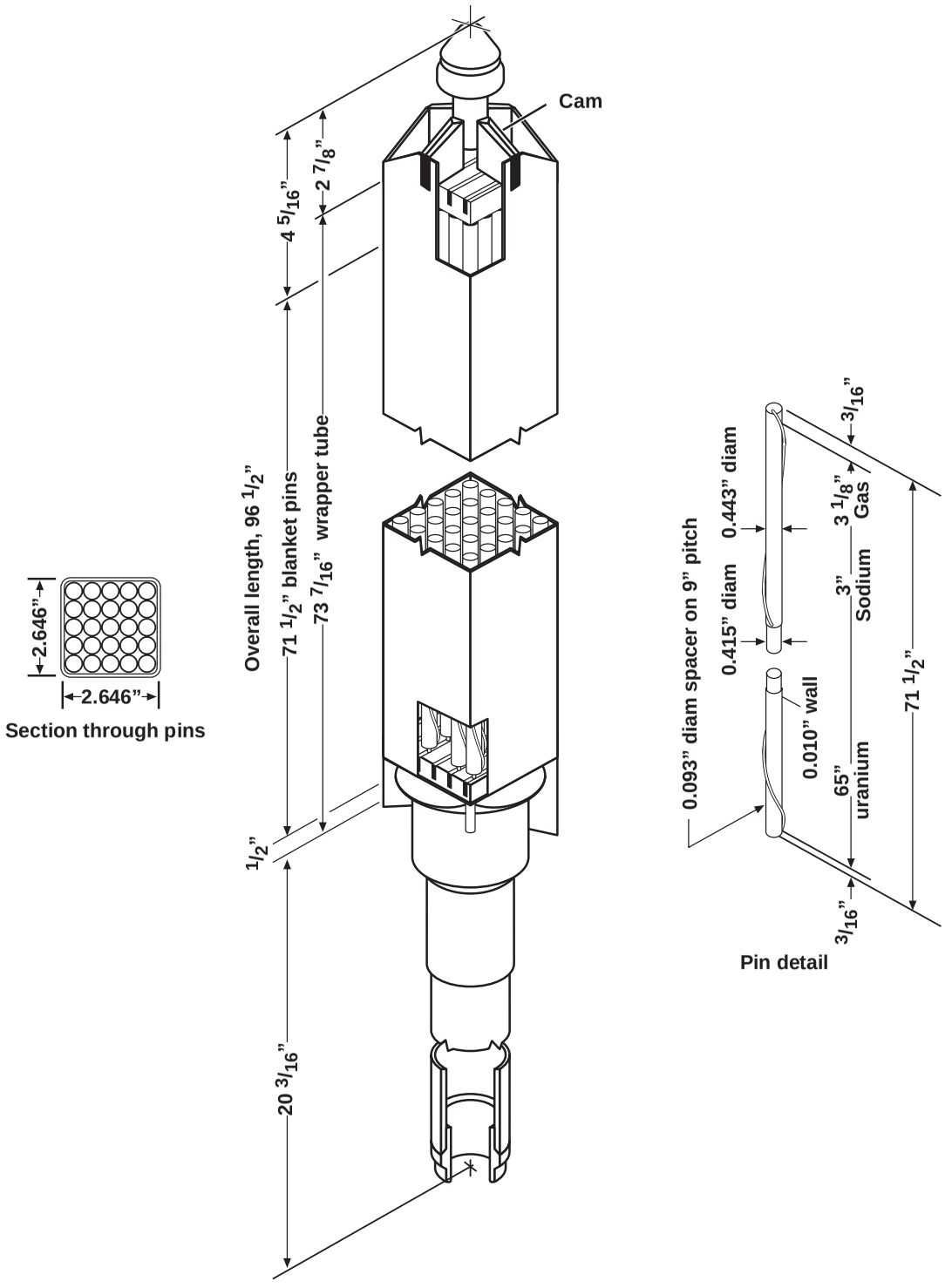


Figure D–2 Fermi-1 Radial Blanket Assembly

Table D–6 Description of Fermi-1 Blanket Elements and Assemblies

Property	*Axial Blanket*	*Radial Blanket*
Element Description:		
Cladding material	Stainless steel 304	Stainless steel 304
Clad outside diameter (inches)	0.443	0.443
Clad thickness (inches)	0.010	0.010
Uranium length (inches)	14	65
Fuel elements (pins or rods) per assembly	16 in upper blanket 16 in lower blanket	25
Subassembly Description:		
Cross-section shape	square	Square
Outside dimension (inches)	2.646	2.646
Wall thickness (inches)	0.096	0.096
Number of assemblies	403[a]	559
General Composition:		
Uranium alloy composition	U–2.75 weight percent Molybdenum	U–2.75 weight percent Molybdenum
Uranium-235 enrichment	0.35 percent	0.35 percent
Sodium (grams/element)	5.5	20.7

[a] Includes both upper and lower axial blankets.

D.4.3 Storage

After the Fermi-1 reactor was permanently shutdown, the blanket assemblies were placed into 14 canisters and transported to INTEC in 1974 and 1975 in 14 shipments. The 14 canisters are made of stainless steel with a carbon steel basket inside. The canisters are 3.4 meters (11 feet, 2.5 inches) long and 65 centimeters (25.5 inches) in diameter. The canisters were filled with helium and seal welded. Twelve of the canisters contain the radial blanket assemblies and two of the canisters contain the shorter axial blanket assemblies.

D.5 Fast Flux Test Facility and Other Miscellaneous Fuels

As shown in Table D–2, the majority of the spent nuclear fuel addressed by this EIS is EBR-II driver, EBR-II blanket, or Fermi-1 blanket. However, there are small quantities of other spent nuclear fuel that also contain metallic sodium that are included in the scope of this EIS. These miscellaneous materials are described below.

D.5.1 Fast Flux Test Facility

Background – The Fast Flux Test Facility, located on the Hanford Site near Richland, Washington in southeastern Washington state, is a 400-megawatt thermal nuclear test reactor cooled by liquid sodium. It was built in 1978 and achieved initial criticality in 1980. The Fast Flux Test Facility was built to test plant equipment and fuel for the U.S. Government's liquid metal reactor development program. Although the facility is not a breeder reactor, this program demonstrated the technology of commercial breeder reactors. It was constructed to verify the safety and optimal performance of the key reactor systems and components. It was also intended to ensure the safety and best design of mixed oxide fuel, a mixture of uranium oxide and plutonium oxide.

The Fast Flux Test Facility successfully tested advanced nuclear fuels, materials, and safety designs. It also produced a large number of different medical isotopes, and made tritium for the U.S. fusion research program. Its operation also demonstrated the reactor's inherent safety features—most notably its ability during an emergency to remove reactor decay (residual) heat without pumps or any other mechanical system, simply

based on its design. By contrast, current conventional water reactors require complex safety cooling and backup systems to remove their decay heat.

The United States eventually abandoned the liquid metal reactor program. Shutdown of the Fast Flux Test Facility was ordered to begin in December 1993 after a panel commissioned by the Secretary of Energy concluded that there was no combination of missions that could sustain the cost of the facility operations over ten years. Work to deactivate the reactor began in January 1994.

Description – Under normal operating conditions of the Fast Flux Test Facility, mixed oxide fuel with an enrichment of 20-30 percent plutonium was fabricated and inserted in the reactor core. However, the Fast Flux Test Facility also tested a number of experimental fuels. The material included in the scope of this EIS is the sodium-bonded experimental fuels that were irradiated. **Table D–7** provides data on the sodium-bonded Fast Flux Test Facility spent nuclear fuel addressed by this EIS.

Table D–7 Description of the Fast Flux Test Facility Sodium-Bonded Spent Nuclear Fuel

Property	*Fast Flux Test Facility Spent Nuclear Fuel*
Element Description:	
Shape	Round rod
Cladding material	Stainless steel 316 Stainless steel D9 Stainless steel HT9
Clad outside diameter (inches)	0.23 to 0.38
Clad thickness (inches)	0.022
Element length (inches)	93 to 120
Fuel pins or rods per assembly	217
Sodium (grams/element)	9 to 40
General Composition:	
Uranium alloy composition	Uranium-10-Zirconium[a] Uranium-10-Plutonium-10-Zirconium Plutonium/Uranium Carbide
Uranium-235 enrichment	0.2 percent to 24 percent
Typical burnup (megawatt days/metric ton uranium)	68,000 to 140,000
Assembly Description:	
Rods per assembly	217
Assembly shape	Hexagon
Assembly width (inches)	4.567 flat to flat
Assembly height (inches)	144

[a] An alloy of 90 weight percent uranium and 10 weight percent zirconium.

Storage – The Fast Flux Test Facility sodium-bonded spent nuclear fuel is currently in dry storage at the facility. The facility has no major vulnerabilities.

Inventory – There are just over 1,600 Fast Flux Text Facility rods (approximately 300 individual rods or elements and 6 assemblies consisting of 217 rods each) which are sodium-bonded totaling 0.32 metric tons of heavy metal. (Of this fuel, 0.07 metric tons of heavy metal, consisting of approximately 100 rods or

elements and 1 assembly, are unirradiated fuel.) The radionuclide inventory of this spent nuclear fuel is presented in Table D–4.

D.5.2 Miscellaneous Fuels

Sandia National Laboratory Experiments

Background – A series of debris bed experiments were conducted at the Sandia National Laboratory's Annular Core Research Reactor from 1977 to 1985. These experiments were part of a program to study the "coolability" of debris beds that might be formed during reactor accidents. In the event of a severe accident in a sodium-cooled fast reactor, molten core materials may interact with liquid sodium and thus result in rapid quenching, freezing, and fragmentation. This fragmented debris may settle on horizontal surfaces within the reactor vessel to form debris beds. If the beds are subcritical, the debris will be heated by the radioactive decay of retained fission products. The possibility of damage to the pressure vessel and the containment, which prevent or mitigate the release of fission products as a consequence of the accident, depends on the extent to which natural cooling of the debris can be relied to remove decay heat from the bed. The debris bed experiments were the first "coolability" experiments to be conducted in-pile, using internally heated uranium dioxide and sodium.

Description – Each experiment consists of either a single or double containment within a helium chamber in the experiment section. Older experiments had a single containment, while newer ones were doubly contained. The uranium dioxide fuel, sodium, thermocouples, and in newer experiments, the insulated crucible are within the inner containment vessel. The uranium dioxide used in the experiments was produced by Los Alamos National Laboratory. The fuel was not irradiated prior to use in these experiments, nor was it melted during the experiments.

Figure D–3 provides a cut-away view of a typical debris bed experiment. As shown, these experiments are considerably different than the arrangement of sodium-bonded spent nuclear fuel. The fuel is just a small portion of the overall experiment structure. The fuel bed is held in a tantalum-tungsten alloy crucible with zirconia insulation. Each of the experiments is 10-centimeters (4-inches) in diameter and 50-centimeters (20-inches) long.

Storage – The 7 debris bed experiments are stored dry at Sandia National Laboratories in Tech Area 5. The experiments are presently stored in 7 "Dense Packs," a set of underground storage holes in Tech Area 5. There are no known vulnerabilities with this storage.

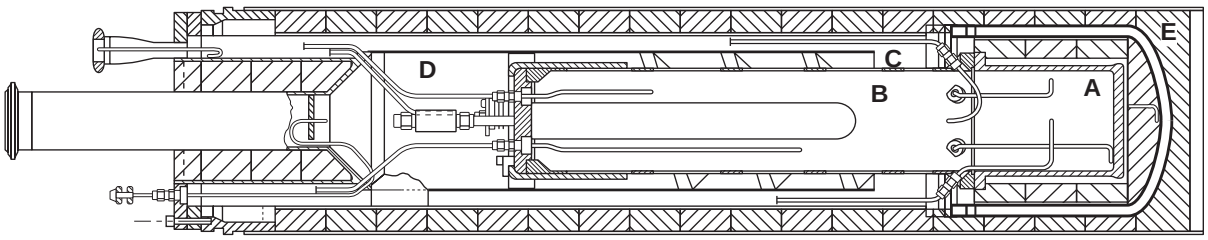
Inventory – The 7 debris bed experiments have a total mass of 650 kilograms (295 pounds), of which only 34 kilograms (15 pounds) is highly-enriched uranium (93 percent Uranium-235) and 20 kilograms (9 pounds) is metallic sodium. The sodium is interdispersed within the fuel debris. The burnup on this spent nuclear fuel is minor since the fuel had not been irradiated prior to these experiments.

The radionuclide inventory for these experiments was modeled as the experimental spent nuclear fuel on a heavy metal basis (see Table D–4). This is considered conservative because of the very low burnup of the experiments and the long cooling time (since 1977 to 1985, depending upon the experiment).

Westinghouse Atomic Power Division

Background – When the Engineering Test Reactor at INEEL was being taken to power, the activity of the primary reactor, water rose abruptly. Within a few minutes after the rise began, the reactor received a slow setback which reduced power. Water chemistry analysis indicated a rupture in an experiment capsule. A small crack was found in one of the Westinghouse Atomic Power Division experiments (WAPD-49-AQ). There

were 15 other similar experiment capsules in the reactor at the time. All of these capsules were removed from the reactor.



A bed in containment vessel
B sodium
C entrance for helium coolant to helical upflow heat exchanger
D exit plenum for hot helium
E typical solid insulation

Figure D–3 Typical Debris Bed Experiment

Description – The capsules have an overall length of 94.6 centimeters (37.25 inches) and are about 12.7 centimeters (5 inches) in diameter. 30 centimeters (12 inches) of each capsule holds the fuel sample assembly. Each fuel sample assembly holds four fuel pins, each having a length of 14 centimeters (5.5 inches) and diameter of 0.9 centimeters (0.34 inches). The fuel pins contain uranium dioxide pellets (18 percent enriched). The oxide pellets have either 1 or 2 sheaths. The sheaths are made of either 304 stainless steel or zircaloy. The fuel pins that have 2 sheaths have a mixture of sodium and potassium between them. **Figure D–4** show the typical Westinghouse Atomic Power Division capsule arrangement.

Storage – The Westinghouse Atomic Power Division spent nuclear fuel is currently stored in INTEC-603. There are a total of 22 experiments (i.e., pins). There are 4 experiments stored each in 5 aluminum cans and 2 capsules in the final can.

Inventory – The total inventory of the Westinghouse Atomic Power Division spent nuclear fuel is 6.6 kilograms (14.5 pounds) of uranium, at 18 percent enrichment. A radionuclide inventory of the Westinghouse Atomic Power Division spent nuclear fuel will be scaled conservatively from the experimental fuel inventory (see Table D–4) based upon heavy metal. This scaling approach is conservative because the experiments are fabricated with plutonium and uranium, have a lower enrichment, and have a lower burnup.

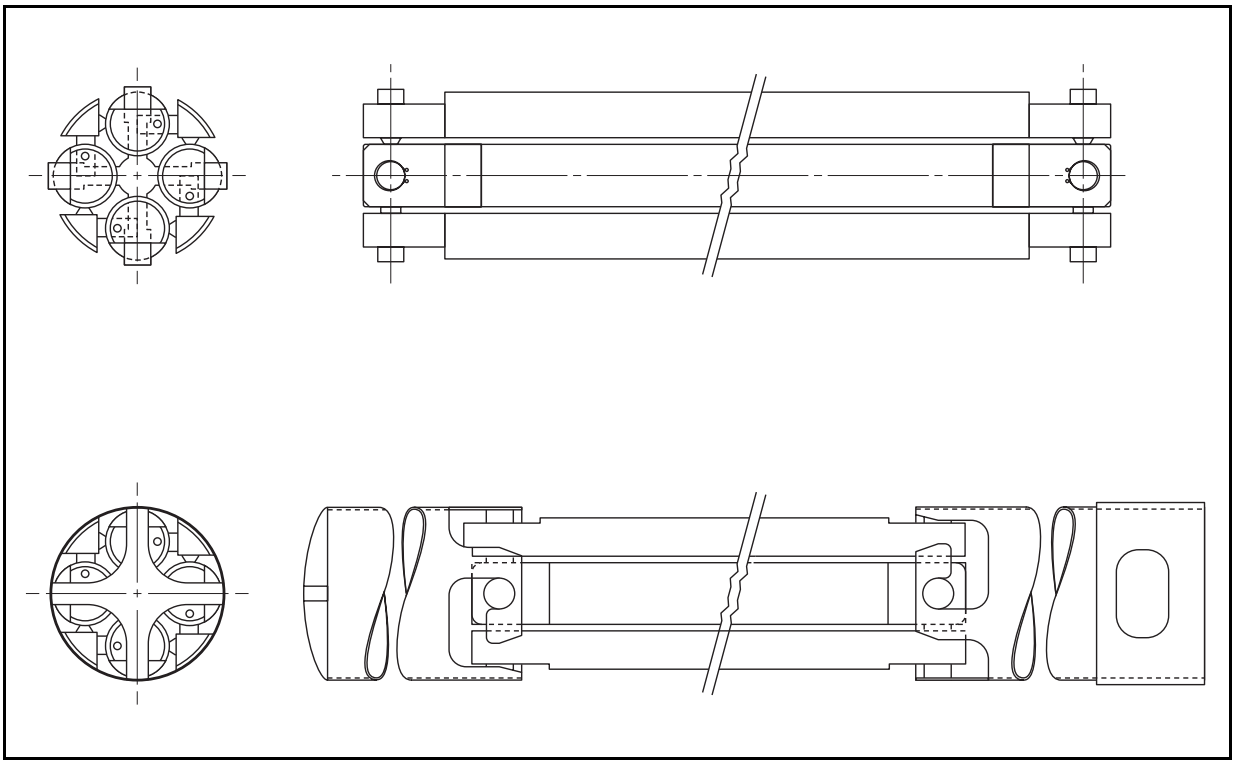


Figure D–4 Diagram of the Westinghouse Atomic Power Division Capsule

Oak Ridge National Laboratory Fast Reactor Spent Nuclear Fuel

Background – On August 12, 1998, the fuel elements were being sheared in half when a "sparkler-like reaction" was observed, lasting less than 30 seconds. This observed reaction was suspected of being an indication of sodium bonding on the spent nuclear fuel. This has not yet been confirmed. This spent nuclear fuel is included in this listing of sodium-bonded spent nuclear fuel in the event that it does prove to be sodium-bonded.

Description – The spent nuclear fuel is considered to be experimental EBR-II spent nuclear fuel elements. They are reported to be a uranium-carbide composition with stainless steel cladding. Figure D–1 shows the general configuration of EBR-II fuel, including experimental fuel. Table D–5 provides data on experimental EBR-II spent nuclear fuel.

Storage – This spent nuclear fuel is currently stored at the Oak Ridge National Laboratory in Building 3525, the Irradiated Fuels Examination Laboratory. The Irradiated Fuels Examination Laboratory is a 2-story brick structure which contains hot cells. Disassembly and examination of fuel and components continue to be the mission of the facility. There are no identified vulnerabilities associated with this facility.

This spent nuclear fuel is stored in 4 containers in Building 3525. The containers are about 1.3 centimeters (0.5 inches) in diameter by 107 centimeters (42 inches) long.

Inventory – This spent nuclear fuel contains a total of 0.38 kilograms (0.84 pounds) of uranium, 0.35 kilograms (0.77 pounds) of which is uranium-235. Therefore, the enrichment is over 90 percent. This spent nuclear fuel also contains a total of 0.091 kilograms (0.20 pounds) of plutonium, 0.084 kilograms (0.18 pounds) of which is plutonium-239 or plutonium-241.

The radionuclide inventory for this small amount of material can be approximated by scaling the experimental spent nuclear fuel inventory (see Table D–4) based on heavy metal. This scaling approach is appropriate since this an EBR-II experimental fuel.

Sodium Research Experiment at SRS

Background – The Sodium Research Experiment was a sodium-cooled, graphite-moderated reactor owned by DOE (and its predecessor AEC) and Southern California Edison, Co. The Sodium Research Experiment achieved initial criticality in 1957 and was last operated in 1964. The Sodium Research Experiment operated at 20 megawatts-thermal until shut down February of 1964 for modification to permit an increase in power level to 30 megawatts-thermal. In December 1966, deactivation was announced.

Description – The Core I Sodium Research Experiment fuel was an unalloyed, uranium metal matrix, with a 2.8 percent uranium-235 enrichment stainless steel type 304 cladding, and sodium-potassium bonding. The Core I fuel contained seven rods per assembly. Core I was removed in 1959 after an incident resulted in the overheating and failure of one or more fuel in a number of fuel assemblies. The 26 undamaged fuel assemblies were shipped to Oak Ridge National Laboratory and were reprocessed. The assemblies that had damaged rods, along with miscellaneous fuel pieces retrieved from the reactor, were packaged into stainless steel canisters.

Core II assemblies were a thorium – 7.6 percent uranium alloy with a 92.3 percent uranium-235 enrichment, stainless steel type 304 cladding and sodium-potassium bonding. Core II fuel contained only five rods per assembly. Each rod contained 12 fuel slugs. Each fuel slug was ¾-inch in diameter and 6 inches long. **Figure D-5** shows the typical assembly. The Core II fuel assemblies were removed from the reactor and placed into storage in 1964. This fuel was declad by Atomics International and shipped to SRS for reprocessing in 1976 and 1977.

In addition to the typical fuel, the Sodium Research Experiment also contained several experimental fuels. The experimental fuel addressed by this EIS is a uranium carbide fuel with a 9.8 percent uranium-235 enrichment, and stainless steel type 304 cladding.

Storage – The uranium carbide spent nuclear fuel addressed by this EIS is currently stored in the Receiving Basin for Offsite Fuels at the SRS. The Sodium Research Experiment spent nuclear fuel is stored in a can 3½ inches in outer diameter and 12 feet long.

Inventory – This spent nuclear fuel contains a total of 43 kilograms (95 pounds) of uranium, 4.2 kilograms (9 pounds) of which is uranium-235. Therefore, the enrichment is 9.8 percent. This spent nuclear fuel also contains a total of 0.016 kilograms (0.035 pounds) of plutonium.

The radionuclide inventory for this small amount of material can be approximated by scaling the experimental spent nuclear fuel inventory (see Table D–4) based on heavy metal. This scaling approach is appropriate since this is a very small quantity of spent nuclear fuel with a burnup lower than the EBR-II spent nuclear fuel.

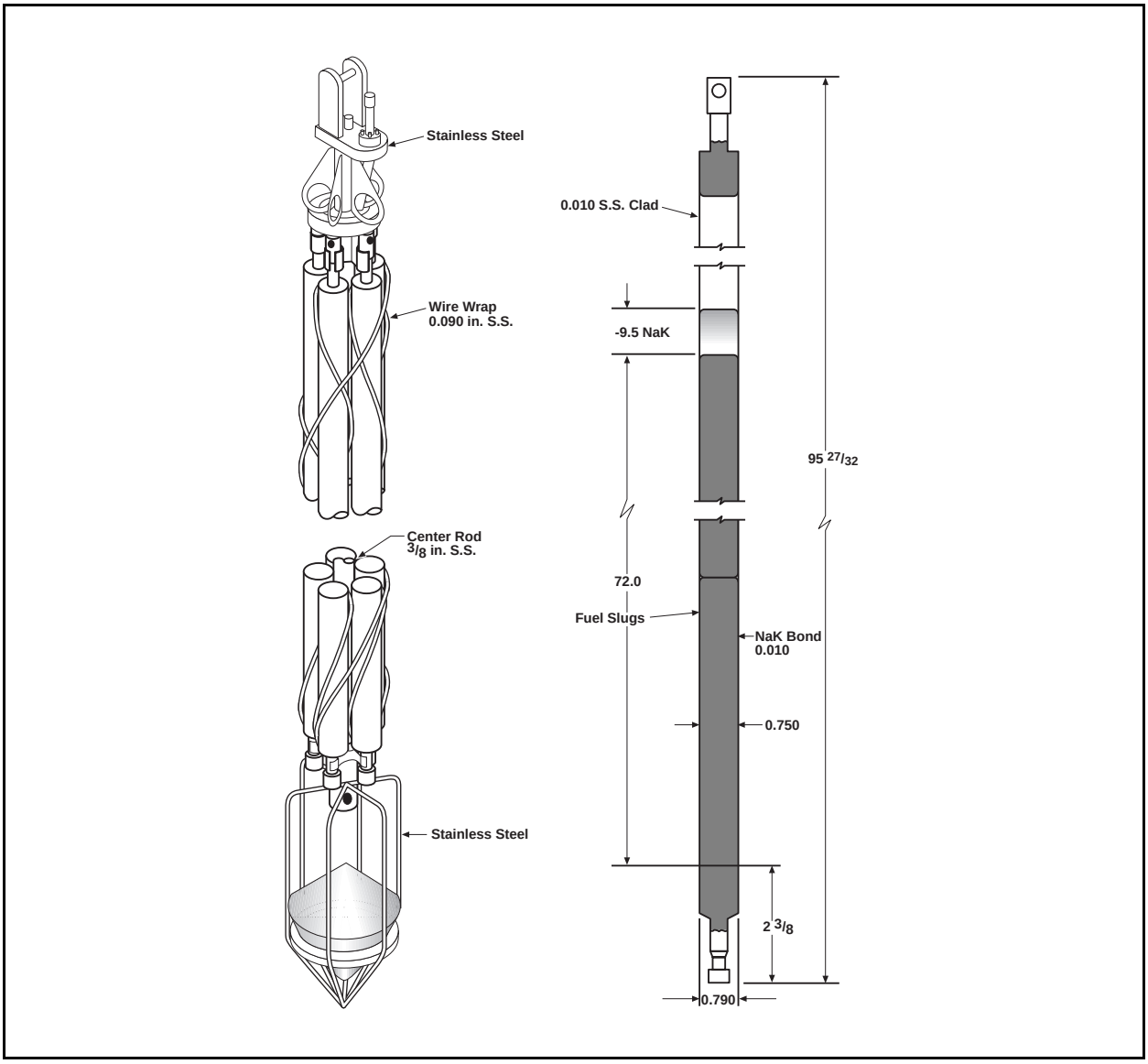


Figure D–5 Sodium Research Experiment Fuel Rod and Assembly Configuration

D.6 REFERENCES

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APPENDIX E
EVALUATION OF HUMAN HEALTH EFFECTS FROM NORMAL OPERATIONS

E.1 INTRODUCTION

This appendix provides a brief general discussion on radiation, and its associated health effects, and describes the method and assumptions used for estimating the potential impacts and risks to individuals and the general public from exposure to the releases of radioactivity and hazardous chemicals during normal operations at the proposed facilities. Information regarding potential radiological impacts resulting from facility accidents is provided in Appendix F of this environmental impact statement (EIS).

This appendix presents numerical information using engineering and/or scientific notation. For example, the number 100,000 can also be expressed as 1×10^5. The fraction 0.00001 can also be expressed as 1×10^{-5}. The following chart defines the equivalent numerical notations that may be used in this appendix.

FRACTIONS AND MULTIPLES OF UNITS

Multiple	*Decimal Equivalent*	*Prefix*	*Symbol*
1×10^6	1,000,000	mega-	M
1×10^3	1,000	kilo-	k
1×10^2	100	hecto-	h
1×10	10	deka-	da
1×10^{-1}	0.1	deci-	d
1×10^{-2}	0.01	centi-	c
1×10^{-3}	0.001	milli-	m
1×10^{-6}	0.000001	micro-	μ
1×10^{-9}	0.000000001	nano-	n
1×10^{-12}	0.000000000001	pico-	p
1×10^{-15}	0.000000000000001	femto-	f
1×10^{-18}	0.000000000000000001	atto-	a

E.2 RADIOLOGICAL IMPACTS ON HUMAN HEALTH

Radiation exposure and its consequences are topics of interest to the general public. For this reason, this EIS places much emphasis on the consequences of exposure to radiation, provides the reader with background information on the nature of radiation, and explains the basic concepts used in the evaluation of radiation health effects.

E.2.1 Nature of Radiation and Its Effects on Humans

What Is Radiation?

Radiation is energy transferred in the form of particles or waves. Globally, human beings are exposed constantly to radiation from the solar system and from the earth's rocks and soil. This radiation contributes to the natural background radiation that always surrounds us. Manmade sources of radiation also exist, including medical and dental x-rays, household smoke detectors, and materials released from nuclear and coal-fired power plants.

All matter in the universe is composed of atoms. Radiation comes from the activity of tiny particles within an atom. An atom consists of a positively charged nucleus (central part of an atom) with a number of negatively charged electron particles in various orbits around the nucleus. There are two types of particles in the nucleus: neutrons that are electrically neutral and protons that are positively charged. Atoms of different types are known as elements. There are more than 100 natural and manmade elements. An element has equal numbers of electrons and protons. When atoms of an element differ in their number of neutrons, they are called isotopes of that element. All elements have three or more isotopes, some or all of which could be unstable (i.e., decay with time).

Unstable isotopes undergo spontaneous change, known as radioactive disintegration or radioactive decay. The process of continuously undergoing spontaneous disintegration is called radioactivity. The radioactivity of a material decreases with time. The time it takes a material to lose half of its original radioactivity is its half-life. An isotope's half-life is a measure of its decay rate. For example, an isotope with a half-life of eight days will lose one-half of its radioactivity in that amount of time. In eight more days, one-half of the remaining radioactivity will be lost, and so on. Each radioactive element has a characteristic half-life. The half-lives of various radioactive elements may vary from millionths of a second to millions of years.

As unstable isotopes change into more stable forms, they emit electrically charged particles. These particles may be either an alpha particle (a helium nucleus) or a beta particle (an electron), with various levels of kinetic energy. Sometimes these particles are emitted in conjunction with gamma rays. The alpha and beta particles are frequently referred to as ionizing radiation. Ionizing radiation refers to the fact that the charged particle energy force can ionize, or electrically charge, an atom by stripping off one of its electrons. Gamma rays, even though they do not carry an electric charge as they pass through an element, can ionize its atoms by ejecting electrons. Thus, they cause ionization indirectly. Ionizing radiation can cause a change in the chemical composition of many things, including living tissue (organs), which can affect the way they function.

When a radioactive isotope of an element emits a particle, it changes to an entirely different element, one that may or may not be radioactive. Eventually, a stable element is formed. This transformation, which may take several steps, is known as a decay chain. For example, radium, which is a member of the radioactive decay chain of uranium, has a half-life of 1,622 years. It emits an alpha particle and becomes radon, a radioactive gas with a half-life of only 3.8 days. Radon decays first to polonium, then through a series of further decay steps to bismuth, and ultimately to lead, which is a stable element. Meanwhile, the decay products will build up and will eventually die away as time progresses.

The characteristics of various forms of ionizing radiation are briefly described below and in the box at right (see Glossary for further definition):

Radiation Type	Typical Travel Distance in Air	Barrier
α	Couple of centimeters	Sheet of paper or skin's surface
β	Few meters	Thin sheet of aluminum foil or glass
γ	Very Large [a]	Thick wall of concrete, lead, or steel
n	Very Large	Water, paraffin, graphite

[a] *Would be infinite in a vacuum*

Alpha (α)

Alpha particles are the heaviest type of ionizing radiation. They can travel only a couple of centimeters in air. Alpha particles lose their energy almost as soon as they collide with anything. They can be stopped easily by a sheet of paper or by the skin's surface.

Beta (β)

Beta particles are much (7,330 times) lighter than alpha particles. They can travel a longer distance than alpha particles in the air. A high energy beta particle can travel a few meters in the air. Beta particles can pass through a sheet of paper, but may be stopped by a thin sheet of aluminum foil or glass.

Gamma (γ)

Gamma rays (and x-rays), unlike alpha or beta particles, are waves of pure energy. Gamma rays travel at the speed of light. Gamma radiation is very penetrating and requires a thick wall of concrete, lead, or steel to stop it.

Neutrons (n)

Neutrons are particles that contribute to radiation exposure both directly and indirectly. The most prolific source of neutrons is a nuclear reactor. Indirect radiation exposure occurs when gamma rays and alpha particles are emitted following neutron capture in matter. A neutron has about one quarter the weight of an alpha particle. It will travel in the air until it is absorbed in another element.

Units of Radiation Measure

During the early days of radiological experience, there was no precise unit of radiation measure. Therefore, a variety of units were used to measure radiation. These units were used to determine the amount, type, and intensity of radiation. Just as heat can be measured in terms of its intensity or effects using units of calories or degrees, amounts of radiation or its effects can be measured in units of curies, radiation absorbed dose (rad), or dose equivalent (rem). The following summarizes those units (see also the definition in the Glossary).

Curie

The curie, named after the French scientists Marie and Pierre Curie, describes the "intensity" of a sample of radioactive material. The rate of decay of 1 gram of radium is the basis of this unit of measure. It is equal to 3.7×10^{10} disintegrations (decays) per second.

Rad

The rad is the unit of measurement for the physical absorption of radiation. The total energy absorbed per unit quantity of tissue is referred to as absorbed dose (or simply dose). As sunlight heats pavement by giving up an amount of energy to it, radiation similarly gives up rads of energy to objects in its path. One rad is equal to the amount of radiation that leads to the deposition of 0.01 joule of energy per kilogram of absorbing material.

Radiation Units and Conversions to International System of Units

1 curie = 3.7×10^{10} becquerel
1 rad = 0.01 gray
1 rem = 0.01 sievert
1 gray = 1 joule/kilogram
1 becquerel = 1 disintegration per second

Rem

A rem is a measurement of the dose equivalent from radiation based on its biological effects. The rem is used in measuring the effects of radiation on the body as degrees centigrade are used in measuring the effects of sunlight heating pavement. Thus, 1 rem of one type of radiation is presumed to have the same biological effects as 1 rem of any other kind of radiation. This allows comparison of the biological effects of radionuclides that emit different types of radiation.

The units of radiation measure in the International Systems of Units are: becquerel (a measure of source intensity [activity]), gray (a measure of absorbed dose), and sievert (a measure of dose equivalent).

An individual may be exposed to ionizing radiation externally (from a radioactive source outside the body) or internally (from ingesting or inhaling radioactive material). The external dose is different from the internal dose because an external dose is delivered only during the actual time of exposure to the external radiation source, but an internal dose continues to be delivered as long as the radioactive source is in the body. The dose from internal exposure is calculated over 50 years following the initial exposure. Both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time.

Sources of Radiation

The average American receives a total of approximately 364 millirem per year from all sources of radiation, both natural and manmade, of which approximately 300 millirem per year are from natural sources (NCRP 1987). The sources of radiation can be divided into 6 different categories: (1) cosmic radiation, (2) terrestrial radiation, (3) internal radiation, (4) consumer products, (5) medical diagnosis and therapy, and (6) other sources (NCRP 1987). These categories are discussed in the following paragraphs.

Cosmic Radiation

Cosmic radiation is ionizing radiation resulting from energetic charged particles from space continuously hitting the earth's atmosphere. These particles and the secondary particles and photons they create comprise cosmic radiation. Because the atmosphere provides some shielding against cosmic radiation, the intensity of this radiation increases with the altitude above sea level. The average dose to people in the United States from this source is approximately 27 millirem per year.

External Terrestrial Radiation

External terrestrial radiation is the radiation emitted from the radioactive materials in the Earth's rocks and soils. The average dose from external terrestrial radiation is approximately 28 millirem per year.

Internal Radiation

Internal radiation results from the human body metabolizing natural radioactive material that has entered the body by inhalation or ingestion. Natural radionuclides in the body include isotopes of uranium, thorium, radium, radon, polonium, bismuth, potassium, rubidium, and carbon. The major contributor to the annual dose equivalent for internal radioactivity are the short-lived decay products of radon, which contribute approximately 200 millirem per year. The average dose from other internal radionuclides is approximately 39 millirem per year.

Consumer Products

Consumer products also contain sources of ionizing radiation. In some products, such as smoke detectors and airport x-ray machines, the radiation source is essential to the products' operation. In other products, such as televisions and tobacco, the radiation occurs as the product's function. The average dose from consumer products is approximately 10 millirem per year.

Medical Diagnosis and Therapy

Radiation is an important diagnostic medical tool and cancer treatment. Diagnostic x-rays result in an average exposure of 39 millirem per year. Nuclear medical procedures result in an average exposure of 14 millirem per year.

Other Sources

There are a few additional sources of radiation that contribute minor doses to individuals in the United States. The dose from nuclear fuel cycle facilities (e.g., uranium mines, mills, and fuel processing plants), nuclear power plants, and transportation routes has been estimated to be less than 1 millirem per year. Radioactive fallout from atmospheric atomic bomb tests, emissions of radioactive material from nuclear facilities, emissions from certain mineral extraction facilities, and transportation of radioactive materials contribute less than 1 millirem per year to the average dose to an individual. Air travel contributes approximately 1 millirem per year to the average dose.

Exposure Pathways

As stated earlier, an individual may be exposed to ionizing radiation both externally and internally. The different ways that could result in radiation exposure to an individual are called exposure pathways. Each type of exposure is discussed separately in the following paragraphs.

External Exposure

External exposure can result from several different pathways, all having in common the fact that the radiation causing the exposure is external to the body. These pathways include exposure to a cloud of radiation passing over the receptor (e.g., an individual member of the public) standing on ground that is contaminated with radioactivity and swimming or boating in contaminated water. If the receptor departs from the source of radiation exposure, the dose rate will be reduced. It is assumed that external exposure occurs uniformly during the year. The appropriate measure of dose is called the effective dose equivalent.

Internal Exposure

Internal exposure results from a radiation source entering the human body through either inhalation of contaminated air or ingestion of contaminated food and water. In contrast to external exposure, once a

radiation source enters the body, it remains there for a period of time that varies depending on decay and biological half-life. The absorbed dose to each organ of the body is calculated for a period 50 years following the intake. The dose equivalent of this absorbed dose is called the committed dose equivalent. Various organs have different susceptibilities to harm from radiation. The quantity that takes these different susceptibilities into account is called the committed effective dose equivalent, and it provides a broad indicator of the risk to the health of an individual from radiation. The committed effective dose equivalent is a weighted sum of the committed dose equivalent in each major organ or tissue. The concept of committed effective dose equivalent applies only to internal pathways.

Radiation Protection Guides

Various organizations have issued radiation protection guides. The responsibilities of the main radiation safety organizations, particularly those that affect policies in the United States, are summarized.

International Commission on Radiological Protection

This commission has the responsibility for providing guidance in matters of radiation safety. The operating policy of this organization is to prepare recommendations to deal with basic principles of radiation protection and to leave to the various national protection committees the responsibility of introducing the detailed technical regulations, recommendations, or codes of practice best suited to the needs of their countries.

National Council on Radiation Protection and Measurements

In the United States, this council is the national organization that has the responsibility to adapt and provide detailed technical guidelines for implementing the International Commission on Radiological Protection recommendations. The organization consists of technical experts who are specialists in radiation protection and scientists who are experts in disciplines that form the basis for radiation protection.

National Research Council/National Academy of Sciences

The National Research Council is an organization within the National Academy of Sciences that associates the broad community of science and technology with the Academy's purposes of furthering knowledge and advising the Federal Government.

Limits of Radiation Exposure

Limits of exposure to members of the public and radiation workers are based on International Commission on Radiological Protection recommendations. Each regulatory organization adopts the International Commission on Radiological Protection's recommendations and sets specific annual exposure limits (usually less than those specified by the commission). The U.S. Department of Energy (DOE) has established a set of limits for radiation workers in 10 CFR 835. **Table E–1** provides the various exposure limits set by the DOE and the U.S. Environmental Protection Agency (EPA) for radiation workers and members of the public.

E.2.2 Health Effects

Radiation exposure and its consequences are topics of interest to the general public. To provide the background for discussions of impacts, this section explains the basic concepts used in the evaluation of radiation effects.

Table E–1 Exposure Limits for Members of the Public and Radiation Workers

Guidance Criteria (Organization)	*Public Exposure Limits at the Site Boundary*	*Worker Exposure Limits*
40 CFR 190 (EPA)	25 millirem per year (all pathways)	—
10 CFR 835 (DOE)	—	5,000 millirem per year
DOE Order N441.1 (DOE)	—	2,000 millirem per year
DOE Order 5400.5 (DOE)	10 millirem per year (all air pathways) 4 millirem per year (drinking water pathway) 100 millirem per year (all pathways)	—
40 CFR 61 (EPA)	10 millirem per year (all air pathways)	—

Radiation can cause a variety of damaging health effects in people. The most significant effects are induced cancer fatalities. These effects are referred to as "latent" cancer fatalities because the cancer may take many years to develop. In the discussions that follow, all fatal cancers are considered latent; therefore, the term "latent" is not used.

The National Research Council's Committee on the Biological Effects of Ionizing Radiation (BEIR) has prepared a series of reports to advise the U.S. Government on the health consequences of radiation exposures. *Health Effects of Exposure to Low Levels of Ionizing Radiation*, BEIR V (National Research Council 1990), provides the most current estimates for excess mortality from leukemia and cancers, other than leukemia, that are expected to result from exposure to ionizing radiation. BEIR V provides estimates that are consistently higher than those in its predecessor, BEIR III. This increase is attributed to several factors, including the use of a linear dose response model for cancers other than leukemia, revised dosimetry for the Japanese atomic bomb survivors, and additional follow-up studies of the atomic bomb survivors and others associated with them. BEIR III employs constant, relative, and absolute risk models, with separate coefficients for each of several sex and age-at-exposure groups. BEIR V develops models in which the excess relative risk is expressed as a function of age at exposure, time after exposure, and sex for each of several cancer categories. The BEIR III models were based on the assumption that absolute risks are comparable between the atomic bomb survivors and the U.S. population. BEIR V models were based on the assumption that the relative risks are comparable. For a disease such as lung cancer, where baseline risks in the United States are much larger than those in Japan, the BEIR V approach leads to larger risk estimates than the BEIR III approach.

The models and risk coefficients in BEIR V were derived through analyses of relevant epidemiologic data that included the Japanese atomic bomb survivors, ankylosis spondylitis patients, Canadian and Massachusetts fluoroscopy (breast cancer) patients, New York postpartum mastitis (breast cancer) patients, Israeli tinea capitis (thyroid cancer) patients, and Rochester thymus (thyroid cancer) patients. Models for leukemia, respiratory cancer, digestive cancer, and other cancers used only the atomic bomb survivor data, although results of analyses of the ankylosis spondylitis patients were considered. Atomic bomb survivor analyses were based on revised dosimetry, with an assumed relative biological effectiveness of 20 for neutrons, and were restricted to doses less than 400 rads. Estimates of risks of fatal cancers, other than leukemia, were obtained by totaling the estimates for breast cancer, respiratory cancer, digestive cancer, and other cancers.

The National Council on Radiation Protection and Measurements (NCRP 1993), based on the radiation risk estimates provided in BEIR V and the International Commission on Radiological Protection Publication 60

recommendations (ICRP 1991), has estimated the total detriment resulting from low dose[1] or low dose rate exposure to ionizing radiation to be 0.00073 per rem for the general population, and 0.00056 per rem for the working population. The total detriment includes fatal and nonfatal cancer and severe hereditary (genetic) effects. The major contribution to the total detriment is from fatal cancer and is estimated to be 0.0004 and 0.0005 per rem for the radiation workers and the general population, respectively. **Table E–2** provides the breakdown of the risk factors for both the workers and the general population. Nonfatal cancers and genetic effects are less probable consequences of radiation exposure. To simplify the presentation of the impacts, estimated effects of radiation are calculated only in terms of latent cancer fatalities.

Table E–2 Nominal Health Effects Coefficients (Risk Factors) from Ionizing Radiation

Exposed Population	*Fatal Cancer* [a,c]	*Nonfatal Cancer* [b]	*Genetic Disorders* [b]	*Total*
Working Population	0.0004	0.00008	0.00008	0.00056
General Population	0.0005	0.0001	0.00013	0.00073

[a] For fatal cancer, the health effect coefficient is the same as the probability coefficient.
[b] In determining a means of assessing health effects from radiation exposure, the International Commission on Radiological Protection has developed a weighting method for nonfatal cancers and genetic effects. Genetic effects only can be applied to a population, not individuals.
[c] For high individual exposures (greater than or equal to 20 rem), the health factors are multiplied by a factor of 2.

Source: NCRP 1993.

The numerical estimates of fatal cancers presented in this EIS were obtained using a linear extrapolation from the nominal risk estimated for lifetime total cancer mortality, which is 0.1 Gray (10 rad). 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 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).

Health Effect Risk Factors Used in This EIS

Health impacts from radiation exposure, whether from sources external or internal to the body, generally are identified as "somatic" (i.e., affecting the exposed individual) or "genetic" (i.e., affecting descendants of the exposed individual). Radiation is more likely to produce somatic effects than genetic effects. The somatic risks of most importance are induced cancers. Except for leukemia, which can have an induction period (time between exposure to carcinogen and cancer diagnosis) of as little as 2 to 7 years, most cancers have an induction period of more than 20 years.

For a uniform irradiation of the body, the incidence of cancer varies among organs and tissues; the thyroid and skin demonstrate a greater sensitivity than other organs. Such cancers, however, also produce relatively low mortality rates because they are relatively amenable to medical treatment. Because fatal cancer is the most probable serious effect of environmental and occupational radiation exposures, estimates of cancer fatalities rather than cancer incidence are presented in this EIS. The numbers of fatal cancers can be used to compare the risks among the various alternatives.

[1] *The low dose is defined as the dose level where DNA repair can occur in a few hours after irradiation-induced damage. Currently, a dose level of about 0.2 Grays (20 rad), or a dose rate of 0.1 milligrays (0.01 rad) per minute is considered to allow the DNA to repair itself in a short period (EPA 1994).*

Based on the preceding discussion and the values presented in Table E–2, the fatal cancers to the general public during normal operations and for accidents in which individual doses are less than 20 rem are calculated using a health risk factor of 0.0005 per person-rem. For workers, a risk factor of 0.0004 excess fatal cancer per person-rem is used. This lower value reflects the absence of children (who are more radiosensitive than adults) in the workforce. Nonfatal cancer and genetic disorders among the public are 20 and 26 percent, respectively, of the fatal cancer risk factor. For workers, the health risk estimators are both 20 percent of the fatal cancer risk factor. These factors are not used in this EIS.

The risk factors are used to calculate the statistical expectation of the effects of exposing a population to radiation. For example, in a population of 100,000 people exposed only to natural background radiation (300 millirem per year), it is expected that about 15 latent cancer fatalities per year of exposure would result from this radiation (100,000 persons × 0.3 rem per year × 0.0005 latent cancer fatalities per person-rem = 15 latent cancer fatalities per year).

Calculations of the number of excess fatal cancers associated with radiation exposure do not always yield whole numbers; calculations may yield numbers less than 1.0, especially in environmental impact applications. For example, if a population of 100,000 were exposed to a total dose of only 0.001 rem per person, the collective dose would be 100 person-rem, and the corresponding estimated number of 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 latent cancer fatalities 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, no person (0 people) would incur a latent fatal cancer from the 0.001 rem dose each member would have received. In a small fraction of the groups, 1 latent cancer fatality would result; in exceptionally few groups, 2 or more latent cancer fatalities would occur. The *average* expected number of deaths over 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.

These same concepts apply to estimating the effects of radiation exposure on a single individual. Consider the effects, for example, of exposure to background radiation over a lifetime. The "number of latent cancer fatalities" corresponding to a single individual's exposure over a (presumed) 72-year lifetime to 0.3 rem per year is 0.011 latent cancer fatalities (1 person × 0.3 rem per year × 72 year × 0.0005 latent cancer fatalities per person-rem = 0.011 latent cancer fatalities).

Again, this is a statistical estimate. That is, the estimated effect of background radiation exposure on the exposed individual would produce a 1.1 percent chance that the individual might incur risk of a latent cancer fatality caused by the exposure over his full lifetime. Presented another way, this method estimates that approximately 1.1 percent of the population might die of cancers induced by background radiation.

E.3 Methodology for Estimating Radiological Impacts

The radiological impacts from normal operation of the facilities were calculated using Version 1.485 of the GENII computer code (PNL 1988). Site-specific input data were used, including location, meteorology, population, and source terms. Section E.3.1 briefly describes GENII and outlines the approach used for normal operations.

E.3.1 GENII Computer Code

The GENII computer model, developed by Pacific Northwest National Laboratory, is an integrated system of various computer modules that analyze environmental contamination resulting from acute or chronic releases to, or initial contamination in, air, water, or soil. The model calculates radiation doses to individuals and populations. The GENII computer model is well documented for assumptions, technical approach, method,

and quality assurance issues (PNL 1988). The GENII computer model has gone through extensive quality assurance and quality control steps, including comparing results from model computations with those from hand calculations and performing internal and external peer reviews. Recommendations given in these reports were incorporated into the final GENII computer model, as appropriate.

For this EIS, only the ENVIN, ENV, and DOSE computer modules were used. The codes are connected through data transfer files. The output of one code is stored in a file that can be used by the next code in the system. The functions of the three GENII computer modules used in this EIS are discussed below.

ENVIN

The ENVIN module of the GENII code controls the reading of input files and organizes the input for optimal use in the environmental transport and exposure module, ENV. The ENVIN code interprets the basic input, reads the basic GENII data libraries and other optional input files, and organizes the input into sequential segments based on radionuclide decay chains.

A standardized file that contains scenario, control, and inventory parameters is used as input to ENVIN. Radionuclide inventories can be entered as functions of releases to air or water, concentrations in basic environmental media (air, soil, or water), or concentrations in foods. If certain atmospheric dispersion options have been selected, this module can generate tables of atmospheric dispersion parameters that will be used in later calculations. If the finite plume air submersion option is requested in addition to the atmospheric dispersion calculations, preliminary energy-dependent finite plume dose factors can be prepared as well. The ENVIN module prepares the data transfer files that are used as input by the ENV module; ENVIN generates the first portion of the calculation documentation—the run input parameters report.

ENV

The ENV module calculates the environmental transfer, uptake, and human exposure to radionuclides that result from the chosen scenario for the user-specified source term. The code reads the input files from ENVIN and then, for each radionuclide chain, sequentially performs the precalculations to establish the conditions at the start of the exposure scenario. Environmental concentrations of radionuclides are established at the beginning of the scenario by assuming decay of preexisting sources, considering biotic transport of existing subsurface contamination, and defining soil contamination from continuing atmospheric or irrigation depositions. For each year of postulated exposure, the code then estimates the air, surface soil, deep soil, groundwater, and surface water concentrations of each radionuclide in the chain. Human exposures and intakes of each radionuclide are calculated for: (1) pathways of external exposure from finite atmospheric plumes; (2) inhalation; (3) external exposure from contaminated soil, sediments, and water; (4) external exposure from special geometries; and (5) internal exposures from consumption of terrestrial foods, aquatic foods, drinking water, animal products, and inadvertent intake of soil. The intermediate information on annual media concentrations and intake rates are written to data transfer files. Although these may be accessed directly, they are usually used as input to the DOSE module of GENII.

DOSE

The DOSE module reads the intake and exposure rates defined by the ENV module and converts the data to radiation dose.

E.3.2 Data and General Assumptions

To perform the dose assessments for this EIS, different types of data were collected and generated. This section discusses the various data that were collected and/or generated, along with the assumptions made (WSRC 1999) for performing the dose assessments in this EIS.

Meteorological Data

The meteorological data used for all normal operational scenarios discussed in this EIS were in the form of joint frequency data files. A joint frequency data file is a table listing the fractions of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The joint frequency data files were based on measurements taken over a period of several years at different heights at each of the management facilities and/or sites.

Population Data

Population distributions were based on the *1990 Census of Population and Housing* data (DOC 1992). Projections were determined for the year 2010 (representative year for operations) for areas within 80 kilometers (50 miles) of the release locations. The site population in 2010, assumed to be representative of the population over the operational period evaluated, was used in the impact assessments. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances up to 80 kilometers (50 miles). The grid was centered at the precise location from which the radionuclides were assumed to be released.

Source Term Data

The source terms used to calculate the impacts of normal operations are provided in Section E.4.

Food Production and Consumption Data

Production and consumption rates used in GENII were those established in the U.S. Nuclear Regulatory Commission Regulatory Guide 1.109 (NRC 1977), for the maximally exposed individual and the general public (average individual), see **Tables E–3 and E–4** for details.

Exposed Population

Dose assessments were performed for both members of the general public and workers for each management facility examined in this EIS. These assessments were made to determine the incremental doses that would be associated with the processing alternatives addressed in this EIS. Incremental doses for members of the public were calculated (via GENII) for two different types of receptors:

- Maximally Exposed Offsite Individual—The maximally exposed individual was assumed to be a hypothetical individual located at a position on the site boundary that would yield the highest impacts during normal operations.

- Population—The general population living within 80 kilometers (50 miles) of the facility in the year 2010.

Table E–3 GENII Usage Parameters for Production and Consumption of Terrestrial Food

	Maximally Exposed Offsite Individual				*General Population*			
Food Type	*Growing Time (days)*	*Yield (kilograms per square meter)*	*Holdup Time (days)*	*Consumption Rate (kilograms per year)*	*Growing Time (days)*	*Yield (kilograms per square meter)*	*Holdup Time (days)*	*Consumption Rate (kilograms per year)*
Leafy Vegetables	90.0	1.5	1.0	30.0	90.0	1.5	14.0	15.0
Root Vegetables	90.0	4.0	5.0	220.0	90.0	4.0	14.0	140.0
Fruit	90.0	2.0	5.0	330.0	90.0	2.0	14.0	64.0
Grains/Cereals	90.0	0.8	180.0	80.0	90.0	0.8	180.0	72.0

Source: PNL 1988, NRC 1977.

Table E–4 GENII Usage Parameters for Production and Consumption of Animal Products

			Animal Stored Feed				*Animal Fresh Forage*			
Food Type	*Human Consumption Rate (kilograms per year)*	*Holdup Time (days)*	*Diet Fraction*	*Growing Time (days)*	*Yield (kilograms per square meter)*	*Storage Time (days)*	*Diet Fraction*	*Growing Time (days)*	*Yield (kilograms per square meter)*	*Storage Time (days)*
Maximally Exposed Offsite Individual										
Beef	80.0	15.0	0.25	90.0	0.80	180.0	0.75	45.0	2.00	100.0
Poultry	18.0	1.0	1.00	90.0	0.80	180.0	–	–	–	–
Milk	270.0	1.0	0.25	45.0	2.00	100.0	0.75	30.0	1.50	0.00
Eggs	30.0	1.0	1.00	90.0	0.80	180.0	–	–	–	–
General Population										
Beef	70.0	34.0	0.25	90.0	0.80	180.0	0.75	45.0	2.00	100.0
Poultry	8.5	34.0	1.0	90.0	0.80	180.0	–	–	–	–
Milk	230.0	3.0	0.25	45.0	2.00	100.0	0.75	30.0	1.50	0.00
Eggs	20.0	18.0	1.0	90.0	0.80	180.0	–	–	–	–

Source: PNL 1988, NRC 1977.

Basic Assumptions

To estimate radiological impacts from normal operations, the following additional assumptions and factors were considered in using GENII:

Radiological airborne gaseous and particulate emissions were assumed to be released to the atmosphere through the plant stacks. See Section E.4 for the specifics at each management facility.

- Ground surfaces were assumed to have no previous deposition of radionuclides.

- The annual external exposure time to the plume and to soil contamination was 0.7 years (16.8 hours per day) for the maximally exposed offsite individual (NRC 1977).

- The annual external exposure time to the plume and to soil contamination was 0.5 years (12 hours per day) for the population (NRC 1977).

- The inhalation exposure time to the plume was 1 year for the maximally exposed individual and general population.

- The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of an adult human.

- A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure; inhalation; and ingestion of food crops and animal products contaminated by deposition of radioactivity from the air.

- Resuspension of particulates was not considered because calculations of dust loading in the atmosphere shows that this pathway is negligible compared to the other pathways.

- Reported release heights were used for atmospheric releases and were assumed to be the effective stack heights. The resultant doses were conservative, as use of the actual stack heights negates plume rise.

- The calculated doses were 50-year committed doses from 1 year of intake.

The exposure, uptake, and usage parameters used in the GENII model for normal operations are provided in Tables E–3, E–4, and **E–5**.

Table E–5 GENII Exposure Parameters to Plumes and Soil Contamination (Normal Operations)

Maximally Exposed Offsite Individual				*General Population*			
External Exposure		*Inhalation of Plume*		*External Exposure*		*Inhalation of Plume*	
Plume (hours)	***Soil Contamination (hours)***	***Exposure Time (hours)***	***Breathing Rate (cubic centimeters per second)***	***Plume (hours)***	***Soil Contamination (hours)***	***Exposure Time (hours)***	***Breathing Rate (cubic centimeters per second)***
6,136	6,136	8,766	270	4,383	4,383	8,766	270

Source: PNL 1988, NRC 1977.

Worker doses associated with the processing alternatives were determined from historical data associated with similar operations. See Section E.4 for details.

E.3.3 Uncertainties

The sequence of analyses performed to generate the radiological impact estimates from normal operation include: (1) selection of normal operational modes, (2) estimation of source terms, (3) estimation of environmental transport and uptake of radionuclides, (4) calculation of radiation doses to exposed individuals, and (5) estimation of health effects. There are uncertainties associated with each of these steps. Uncertainties exist in the way the physical systems being analyzed are represented by the computational models and in the data required to exercise the models (due to measurement, sampling, or natural variability).

In principle, one can estimate the uncertainty associated with each source and predict the remaining uncertainty in the results of each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final results. However, conducting such a full-scale quantitative uncertainty analysis is neither practical nor a standard practice for a study of this type. Instead, the analysis is designed to ensure—through judicious selection of release scenarios, models, and parameters—that the results represent the potential risks. This is accomplished by making conservative assumptions in the calculations at each step. The models, parameters, and release scenarios used in the calculations are selected

in such a way that most intermediate results and, consequently, the final estimates of impacts, are greater than would be expected. As a result, even though the range of uncertainty in a quantity might be large, the value calculated for the quantity would be close to one of the extremes in the range of possible values, so the chance of the actual quantity being greater than the calculated value would be low (or the chance of the quantity being less than the calculated value if the criteria are such that the quantity has to be maximized). The goal of the radiological assessment for normal operation in this study has been to produce results that are conservative.

The degree of conservatism in the calculated results is closely related to the range of possible values the quantity can have. This range is determined by what can be expected to realistically occur. Thus, the only processes considered are those that are credible for the conditions under which the physical system being modeled operates. This consideration has been employed for the normal operation analyses.

Although the radionuclide composition of source terms are reasonable estimates, there are uncertainties in the radionuclide inventory and release reactions that affect estimated impacts.

E.4 RADIOLOGICAL RELEASES TO THE ENVIRONMENT AND ASSOCIATED IMPACTS

This section summarizes the estimated radiological releases to the environment as well as resulting impacts associated with the various alternatives assessed in this EIS. Impacts to workers from these alternatives are also discussed. The methodology for estimating radiological impacts, as well as associated input data and analytical assumptions are provided in Section E.3.

E.4.1 Electrometallurgical Treatment of Both Blanket and Driver Fuels (Alternative 1)

Under this alternative, releases of radioactive material would occur during normal operational processing of the sodium-bonded fuel rods in the argon cell at the Fuel Conditioning Facility. Fuel assemblies would be disassembled in the Fuel Conditioning Facility air cell, and individual fuel elements would then be transferred to the argon cell for chopping and treatment in one of the electrorefiners. The entire inventory of gaseous fission products, mainly tritium (H-3) and krypton-85 (Kr-85) is assumed to be released during processing in the Fuel Conditioning Facility. The likelihood of release of radionuclides other than the gaseous fission products is judged to be very small. No radionuclides would be released from the packaged salt and packaged metal waste material transferred from the Fuel Conditioning Facility to the Hot Fuel Examination Facility.

Estimated radioactive releases during normal operations at ANL-W were calculated using a conservative methodology. First, assumptions were made to estimate a maximum annual throughput of material to be processed at the Fuel Conditioning Facility. There would be two electrorefiners in the Fuel Conditioning Facility argon cell; blanket material would be treated in one of the two electrorefiners and driver material would be treated in the other. Both driver and blanket material could be processed each year. Based on an annual operational processing limit of 5,000 kilograms (11,023 pounds) of total heavy metal fuel material consisting of no more than 600 kilograms (1,320 pounds) of heavy metal driver material, it was assumed that driver fuels would be processed at the maximum rate until all driver fuel was processed. In addition, it was assumed that EBR-II fuel (driver and blanket) currently at ANL-W would be processed first. Using these assumptions, annual mass processing throughputs were developed for the purposes of estimating releases of radioactive material during normal operations, and are presented in **Table E–6**.

Table E–6 Annual Processing Assumptions for Estimation of Air Releases of Radionuclides During Normal Operations for Alternative 1 at ANL-W

Year of processing	Driver Fuel (kilograms per year)		Blanket Fuel (kilograms per year)		Total Fuel (kilograms per year)		
	EBR-II[a]	Fast Flux Test Facility[b]	EBR-II[c]	Fermi-1	Driver	Blanket	Driver + Blanket
1	6.0 E+02	0	4.4 E+03	0	6.0 E+02	4.4 E+03	5.0 E+03
2	6.0 E+02	0	4.4 E+03	0	6.0 E+02	4.4 E+03	5.0 E+03
3	6.0 E+02	0	4.4 E+03	0	6.0 E+02	4.4 E+03	5.0 E+03
4	6.0 E+02	0	4.4 E+03	0	6.0 E+02	4.4 E+03	5.0 E+03
5	6.0 E+02	0	4.4 E+03	0	6.0 E+02	4.4 E+03	5.0 E+03
6	1.0 E+02	2.9 E+02	4.0 E+02	4.2 E+03	3.9 E+02	4.6 E+03	5.0 E+03
7	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
8	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
9	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
10	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
11	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
12	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
Totals (kg)	3.10 E+03	2.90 E+02	2.24 E+04	3.42 E+04	3.39 E+03	5.66 E+04	6.00 E+04

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

[a] EBR-II driver consists of 1,100 kilograms of EBR-II driver fuel at ANL-W and 2,000 kilograms at INTEC.

[b] The Fast Flux Test Facility driver consists of 250 kilograms of sodium-bonded Fast Flux Test Facility driver fuel at Hanford, plus 40 kilograms of miscellaneous fuel at INTEC, Sandia National Laboratory, and Oak Ridge Reservation.

[c] EBR-II blanket consists of EBR-II blanket fuel at ANL-W.

Radioactive releases from the Fuel Conditioning Facility argon cell during fuel treatment were estimated next. Radioactivity associated with the fuel to be processed was determined using the fuel radioactivity inventory values discussed in Appendix D. Estimated releases were based on a methodology developed in support of ANL-W's State of Idaho and National Emission Standards for Hazardous Air Pollutants air permitting activities, and agreed upon by the State of Idaho's Department of Environmental Quality (Bauer 1992). From this methodology, equilibrium concentrations in the argon cell (curies per cubic meter per curie processed) were calculated and applied to the inventory associated with the assumed annual throughputs shown in Table E–6. Annual radioactive releases to the atmosphere were calculated as the product of the radionuclide equilibrium concentrations in the argon cell, the annual argon cell atmosphere exhaust (74,400 cubic meters per year), and a conservative adjustment (0.00001) to account for the combined filtration of the two banks of high-efficiency particulate air filters that the cell exhaust must pass through before entering the environment. This filtration adjustment was not applied to tritium or krypton-85, as 100 percent of these radionuclides were assumed to be released.

The Fuel Conditioning Facility stack was modeled with an effective stack height of 60.96 meters. This is the actual stack height, and for conservatism, no plume rise was included in the atmospheric dispersion modeling.

The dose resulting from the release of tritium (H-3) depends heavily on its chemical form. The inhalation dose from oxidized tritium (HTO or T_2O) is 25,000 times higher than for tritium in elemental form (HT or T_2) (ICRP 1982). The dose conversion factors used in the GENII code assume that tritium released to the environment is in the oxidized form and are therefore very conservative for releases that involve elemental tritium. Because of the argon atmosphere in the Fuel Conditioning Facility argon cell, releases of tritium to the cell atmosphere would not become oxidized, and stack releases of tritium would most likely be in the elemental form. The oxidation of elemental tritium to HTO or T_2O has been shown to occur slowly in the environment, and the long-term dose from elemental tritium releases is conservatively estimated to be 1 percent

of that for the oxidized form for this EIS (DOE 1997). Therefore, the inventory of tritium for each year of electrometallurgical treatment processing at the Fuel Conditioning Facility was multiplied by a factor of 0.01 to convert them to an equivalent release of tritium oxide for use as input to the GENII code.

Radiological Gaseous Emissions:

The estimated annual and total atmospheric releases are tabulated in **Table E–7**. This table lists only those radionuclides that resulted from a screening procedure to indicate potential significant dose contribution. The source term listed in Table E–7 for each of the first 5 years of processing (years 1 - 5) represents the source term that results in the highest annual offsite dose, and is therefore used for the maximum annual dose calculations. The project lifetime total values in Table E–7 represent the total estimated releases over the 12 years of processing at ANL-W.

Table E–7 Annual and Total Radioactive Releases During Normal Operations for Alternative 1

	Annual Releases (curies per year)			
Isotope [a]	***Years 1-5***	***Year 6***	***Years 7-12***	***Project Lifetime Total (curies)***
H-3	7.7E+02	6.8E+02	3.8E-01	4.5E+03
C-14	5.2E-07	8.7E-08	2.3E-16	2.7E-06
Fe-55	1.4E-08	2.4E-08	5.8E-13	9.6E-08
Ni-63	6.5E-10	1.7E-10	1.0E-12	3.4E-09
Kr-85	1.2E+04	8.8E+03	3.3E+00	6.7E+04
Sr-90	7.0E-08	5.2E-08	4.7E-11	4.0E-07
Y-90	7.0E-08	5.2E-08	4.7E-11	4.0E-07
Ru-106	3.2E-08	2.9E-08	7.6E-17	1.9E-07
Rh-106	3.2E-08	2.9E-08	7.6E-17	1.9E-07
Cd-113m	6.7E-10	5.2E-10	3.1E-13	3.9E-09
Sb-125	4.1E-08	3.5E-08	3.2E-13	2.4E-07
Te-125m	4.5E-10	3.9E-10	4.0E-15	2.6E-09
I-129	1.4E-12	9.7E-13	1.8E-15	8.2E-12
Cs-134	3.2E-08	4.0E-08	9.5E-16	2.0E-07
Cs-137	4.0E-06	2.9E-06	3.5E-09	2.3E-05
Ba-137m	3.8E-06	2.8E-06	3.3E-09	2.2E-05
Ce-144	1.2E-09	1.8E-09	1.9E-20	7.7E-09
Pr-144	1.2E-09	1.8E-09	1.9E-20	7.7E-09
Pm-147	2.9E-03	2.6E-03	2.3E-08	1.7E-02
Sm-151	2.1E-09	1.4E-09	3.7E-12	1.2E-08
Eu-154	2.1E-10	2.0E-10	2.2E-15	1.3E-09
Eu-155	1.4E-09	1.1E-09	1.9E-13	8.3E-09
Th-228	1.6E-14	1.3E-14	3.2E-19	9.1E-14
U-234	1.2E-11	7.8E-12	7.8E-17	6.7E-11
U-235	3.9E-13	2.6E-13	1.8E-14	2.3E-12
U-236	3.7E-13	2.6E-13	2.7E-16	2.1E-12
U-238	7.4E-13	7.7E-13	8.1E-13	9.4E-12
Np-237	3.2E-13	2.7E-13	2.1E-15	1.9E-12
Pu-238	2.9E-10	2.2E-10	3.4E-14	1.6E-09
Pu-239	7.1E-09	1.2E-09	1.4E-10	3.7E-08

Isotope [a]	Annual Releases (curies per year)			Project Lifetime Total (curies)
	Years 1-5	Year 6	Years 7-12	
Pu-240	4.7E-10	1.2E-10	1.1E-13	2.5E-09
Pu-241	1.9E-09	1.1E-09	3.6E-15	1.1E-08
Am-241	6.2E-12	1.8E-12	1.5E-17	3.3E-11
Am-242m	6.4E-14	9.3E-15	2.6E-24	3.3E-13
Totals	1.2E+04	9.5E+03	3.7E+00	7.1E+04

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

[a] The listed isotopes are present within the argon cell at the Fuel Conditioning Facility. Due to lack (scarcity) of oxygen in the argon cell, the tritium released to the cell would be in molecular (elemental) form (i.e., T_2, or HT).

Population Impacts:

The estimated annual radiological impacts due to the source term for the maximally exposed offsite individual and the 80-kilometer (50-mile) population surrounding ANL-W are tabulated in **Table E–8**. Calculated impacts are shown for each year of processing as well as for each of the fuel types to be processed. Impacts resulting from releases during processing EBR-II driver and EBR-II blanket during each of the first 5 years are listed (years 1 - 5), for processing some of all 4 fuel types during the 6th year (year 6), and for processing Fermi-1 blanket fuel during each of the final 6 years (years 7 - 12). The impacts to the maximally exposed offsite individual and the surrounding population result primarily from estimated releases of tritium and krypton-85. Together, these two radionuclides account for greater than 99.9 percent of the estimated impacts.

Table E–8 Annual Radiological Impacts to the Public From Operational Activities Associated With Alternative 1 at ANL-W

Year(s) of Processing	Fuel Type	Offsite Population		Maximally Exposed Offsite Individual	
		Collective Dose (person-rem per year)	Latent Cancer Fatalities (number of cancers)	Annual Dose (millirem per year)	Latent Cancer Fatality Risk
1 - 5	EBR-II Driver	2.8 E-03	1.4 E-06	3.3 E-04	1.6 E-10
	FFTF Driver	0	0	0	0
	EBR-II Blanket	8.4 E-05	4.2 E-08	1.0 E-05	5.0 E-12
	Fermi Blanket	0	0	0	0
	All Fuels, Years 1 - 5	2.9E-03	1.4 E-06	3.4E-04	1.7 E-10
6	EBR-II Driver	4.6 E-04	2.3 E-07	5.4 E-05	2.7 E-11
	FFTF Driver	1.8 E-03	9.2 E-07	2.2 E-04	1.1 E-10
	EBR-II Blanket	7.6 E-06	3.8 E-09	9.2 E-07	4.6 E-13
	Fermi Blanket	9.1 E-07	4.6 E-10	1.1 E-07	5.5 E-14
	All Fuels, Year 6	2.3 E-03	1.2 E-06	2.8 E-04	1.4 E-10
7 - 12	EBR-II Driver	0	0	0	0
	FFTF Driver	0	0	0	0
	EBR-II Blanket	0	0	0	0
	Fermi Blanket	1.1E-06	5.4 E-10	1.3 E-07	6.5 E-14
	All Fuels, Years 7-12	1.1E-06	5.4 E-10	1.3 E-07	6.5 E-14

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Total cumulative radiological impacts over the projected 13 years of operations under this alternative are tabulated in **Table E–9**. This table shows the sum of the calculated impacts to the maximally exposed offsite individual and the surrounding population for each of the 12 years of processing.

Table E–9 Cumulative Maximum Radiological Impacts From Operational Activities Associated With Alternative 1 at ANL-W

	80-km Offsite Population		*Maximally Exposed Offsite Individual*	
	Collective Dose (person-rem)	***Latent Cancer Fatalities (number of cancers)***	***Dose (millirem)***	***Latent Cancer Fatality Risk***
Total Project Impacts[a]	1.68-02	8.4E-06	2.0 E-03	9.8 E-10

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
[a] Total impacts are estimated for the 12-year duration of fuel treatment.

Worker Impacts:

Workers involved with electrometallurgical treatment activities at ANL-W could receive radiation doses during handling activities, such as receiving and unloading fuel casks, and transferring in-process waste material from the Fuel Conditioning Facility to the Hot Fuel Examination Facility. Doses received during in-cell activities would likely be very small. A maximally exposed worker dose estimate for this EIS is based on the regulatory limit of 5,000 millirem per year for radiation workers at DOE sites. If an individual worker received this dose each year of the 13 years of the electrometallurgical treatment project, the total maximally exposed worker dose would be 65,000 millirem, with an associated risk of 0.026 latent cancer fatalities.

However, actual worker doses are likely to be much lower than this maximum estimate. The ANL-W radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. In addition, ANL-W has established an administrative goal of 1,500 millirem per year to any individual. The general design goals at the Fuel Conditioning Facility, for example, were to maintain radiation fields below 0.5 millirem per hour at all workstations. This means that for an individual working at the Fuel Conditioning Facility for a full-time occupational work year of 2,000 hours, the annual dose would be 1,000 millirem.

Worker population doses were estimated by examining the type and duration of various operations performed by workers involved with the electrometallurgical treatment project. Doses can be estimated based on previous doses from similar activities at ANL-W. Based on information from ANL-W, the total worker population dose estimate is 22 person-rem per year, averaging out to an individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates are extended out over the 13 years of operational activities (12 years of fuel treatment and a final year of high level radioactive waste conversion activities), the cumulative worker population dose is 286 person-rem and the associated risk is 0.11 latent cancer fatalities. The estimated population impacts to the workers associated with this alternative are summarized in **Table E–10**.

Table E–10 Annual and Total Impacts to Electrometallurgical Treatment Workers at ANL-W

	Worker Population	
	Collective Dose (person-rem)	***Latent Cancer Fatalities***
Annual Impacts	2.2 E+01	8.8 E-03
Total Project Impacts[a]	2.9 E+02	1.1 E-01

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
[a] Total impacts are estimated for the 13-year duration of processing.

E.4.2 Blanket Fuel Preparation and Electrometallurgical Treatment of Driver Fuel at ANL-W (Alternatives 2 Through 5)

In Alternatives 2 through 5, the blanket fuel assemblies will need to be prepared at the ANL-W facilities prior to packaging in high-integrity cans or processing in either the PUREX process at SRS or the melt and dilute process at SRS or ANL-W. When the blanket fuel is to be processed at SRS, Alternative 3 (PUREX processing) and 5 (melt and dilute processing), the blanket fuel will be declad and cleaned at ANL-W in the argon cell of the Hot Fuel Examination Facility. Processing of the blanket fuel assemblies at ANL-W (Alternative 2, placing the blanket fuel in high-integrity cans, and Alternatives 4 and 6, melt and dilute) do not require decladding of the blanket fuel. This activity would also be performed in the argon cell of the Hot Fuel Examination Facility. The preparation of the blanket fuel under these alternatives require only that the fuel be cut into segments and cleaned (see Appendix C for details). The following discussion addresses the radiological impact of normal operations at ANL-W for the preparation of the blanket fuel elements and the electrometallurgical treatment of the driver fuel elements. This analysis is applicable to Alternatives 2 through 5.

Gaseous Emissions:

Blanket fuel preparation would occur at the Hot Fuel Examination Facility. These activities would cause gaseous fission products to be released into the argon cell. As stated earlier in Section E.4.1, krypton-85 and elemental tritium are the most prevalent gaseous radionuclides that would be released to the environment. The released tritium 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 the krypton-85 and elemental tritium. The maximum released curies of radioactive gaseous emissions occurs when preparation of the blanket fuel and chopping of the driver fuel (for electrometallurgical treatment processing) are performed simultaneously. This simultaneous operation was estimated to occur over a 6-year period starting in 2003. Based on a blanket fuel preparation throughput of 10 metric tons of heavy metal and an electrometallurgical treatment process rate of about 0.6 metric tons of heavy metal of driver fuel elements annually, about 809 curies of elemental tritium and 11,860 curies of krypton-85 would be released to the atmosphere annually, see **Table E–11**. This release rate would last about 2 years, or until all of the EBR-II blanket fuels are processed, then afterward the release rate would drop during the processing of the Fermi-1 blanket fuels (The release rate for the processing of 10 metric tons of heavy metal of Fermi-1 blanket fuel is less than 1 curie of elemental tritium and 6.6 curies of krypton-85.)

Table E–11 Maximum Annual Radiological Gaseous Emission From Activities Associated With Alternative 2 Through 5 at ANL-W

	Facility	*Maximum Processing Rate (metric tons of heavy metal per year)*	*Duration (years)*	*Annual Release (curies)*	
				Tritium	*Krypton-85*
Driver Fuel	Fuel Conditioning Facility	0.6	6	738	11,340
Blanket Fuel	Hot Fuel Examination Facility	10	6	71.2	520

Population Impacts:

The doses to the maximally exposed offsite individual and the general public residing within the 80 kilometers (50 miles) surrounding ANL-W are presented in **Table E–12**. As stated in Section E.4.1 the dose resulting from the release of tritium is highly dependent upon its chemical form. The doses presented in Table E–12 result from releases that are assumed to be one percent oxidized tritium, the same assumption used in the

analysis of Alternative 1. These impacts are calculated for the preparation of the blanket fuel assemblies, for the processing of the driver fuel assemblies using the EMT process, and the total impacts. These total impacts are applicable to the processing of blanket and driver materials under Alternatives 2, 3, 4, and 5.

Table E–12 Maximum Annual Radiological Impacts to the Public From Operational Activities Associated With Alternatives 2 Through 5 at ANL-W

	Offsite Population		*Maximally Exposed Offsite Individual*	
	Dose (person-rem)	***Latent Cancer Fatalities***	***Dose (millirem)***	***Latent Cancer Fatality Risk***
Driver Fuel	0.0028	1.4×10^{-6}	0.00033	1.6×10^{-10}
Blanket Fuel	0.00028	1.4×10^{-7}	0.000048	2.4×10^{-11}
Total	0.0031	1.6×10^{-6}	0.00038	1.9×10^{-10}

Worker Impacts:

The annual worker dose and worker population dose would be similar to those provided in Section E.4.1.

E.4.3 PUREX Processing at SRS (Alternative 3)

PUREX processing at F-Canyon would release radioactive gaseous fission products during treatment of about 57 metric tons of heavy metal of EBR-II and Fermi-1 blanket fuels. Since declad and cleaned blanket fuels are packaged and sent to SRS, no additional gaseous fission products are expected to be present in that fuel. However, for conservatism, it was assumed that the gaseous fission products in the blanket fuels would remain within the fuel matrix and would be released to the environment during PUREX processing at SRS. As a result, there would be incurred doses to the public associated with PUREX operations. The duration of PUREX operations was estimated to be about 6 months, based on the F-Canyon's throughput and consistent with assumptions made for the treatment duration of a similar type fuel at SRS in the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998).

Gaseous Emissions:

According to SRS Spent Nuclear Fuel EIS data (DOE 1997), tritium and krypton-85 are the only isotopes that would be expected to be released during PUREX processing operations. Based on the assumption that the entire fission gas inventory would remain within the fuel matrix after the decladding and cleaning process, it was assumed that the inventory of krypton-85 and tritium would be released. Using the gaseous fission product inventory provided in Appendix D for the EBR-II and Fermi-1 blanket fuels, the potential airborne radiological release quantities were estimated and presented in **Table E–13**. This inventory was used to calculate the population doses from air emissions.

Liquid Effluent:

PUREX processing is the only process among the alternatives considered that would release measurable radioactive nuclides to the surface water. This release occurs through the cooling water system. The expected radiological effluents from processing of declad and cleaned blanket fuels at F-Canyon were 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 are 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 6-month operation of F-Canyon. Table E–13 lists the radionuclides and their corresponding curies that are estimated to be released during PUREX processing of blanket fuels.

Table E–13 Estimated Incremental Releases of Radiological Air Emissions and Liquid Effluent During Normal Operations of PUREX Processing at F-Canyon

Isotope	*Releases to Air (curies)*	*Releases to Liquid (curies)* [a]
H-3	162.1	1.54
Kr-85	1,187.5	-
Sr-89/90	-	3.13×10^{-5}
Cs-137	-	2.17×10^{-3}
U-234	-	8.48×10^{-5}
U-235	-	1.10×10^{-5}
U-238	-	1.93×10^{-4}
Pu-238	-	1.56×10^{-5}
Pu-239	-	7.76×10^{-6}

[a] Estimated curies using the information provided in Arnett and Mamatey 1998.

Population Impacts:

Estimated annual radiological impacts associated with the F-Canyon PUREX operations for the maximally exposed individual and the general population residing within the 80-kilometer (50-mile) population surrounding F-Canyon are presented in **Table E–14**. This table provides the radiological doses to the public from air emissions and liquid effluents separately. According to SRS analytical assumptions (WSRC 1998), a maximally exposed individual associated with liquid releases is an individual who lives downriver of SRS 365 days per year, drinks 2 liters of untreated water per day from the Savannah River, consumes a large amount of Savannah River fish, and spends the majority of time on or near the river. The general population liquid effluent dose is calculated for the discrete population groups at Beaufort-Jasper and Port Wentworth, as well as for other diffuse population groups that make use of the Savannah River; the majority of this dose is due to the drinking water pathway.

Table E–14 Incremental Radiological Impacts to Public from Normal Operational Releases at F-Canyon During PUREX Processing

Offsite Population				*Maximally Exposed Offsite Individual*			
Air Dose (person-rem)	*Liquid Dose* [a] *(person-rem)*	*Total Dose (person-rem)*	*Latent Cancer Fatalities*	*Air Dose (millirem)*	*Liquid Dose* [a] *(millirem)*	*Total Dose (millirem)*	*Latent Cancer Fatality Risk*
0.019	0.00068	0.020	0.000010	0.00039	0.00012	0.00051	2.6×10^{-10}

[a] The dose values were estimated based on the results presented in DOE 1995 for processing a similar fuel.

For conservatism, as well as being able to show compliance with DOE Order 5400.5 (100 millirem annual dose limit to an individual from all pathways), the incremental airborne and liquid doses associated with the F-Canyon processing were summed together even though it is two distinct individuals who receive a maximum airborne and maximum liquid dose. In addition, for analysis purposes, it was assumed that tritium would be released to the atmosphere in oxide form. The public impacts from radiological liquid effluent were estimated based on the results provided in the SRS's *Interim Management of Nuclear Materials Environmental Impact Statement* (DOE 1995). This is consistent with the approach used in the recent SRS *Spent Nuclear Fuel Management Draft Environmental Impact Statement* (DOE 1998). The SRS Spent Nuclear Fuel management EIS (DOE 1998) used "per unit" values (per metric tons of fuels processed) to estimate liquid doses associated with the PUREX processing of 20 metric tons of heavy metal of declad blanket fuel. This EIS uses the same

approach to estimate the radiological doses to the public from potential radiological liquid effluent from PUREX processing.

Worker Impacts:

Worker impacts were estimated by examining historical doses associated with PUREX processing at SRS; these values were based on those presented in the SRS Spent Nuclear Fuel Management Environmental Impact Statement (DOE 1998). The SRS radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. Doses and associated impacts are based on a 6-month processing period. **Table E–15** presents estimated values to both the average worker and entire workforce population.

Table E–15 Incremental Worker Radiological Impacts Due to Normal Operations at F-Canyon

Worker Population		*Individual Worker*	
Collective Dose (person-rem/yr)	***Latent Cancer Fatalities from 6 Months of Processing***	***Individual Dose (millirem per year)***	***Latent Cancer Fatality Risk from 6 Months of Processing***
75[a]	0.015	500[a]	1.0×10^{-4}

[a] Processing of blanket fuel will require six months of F-Canyon operation, yielding half of the annual doses presented.

E.4.4 Melt and Dilute at SRS Building 105-L Radiological Releases and Impacts (Alternative 5)

Melt and dilute processing at Building 105-L would release radioactive gaseous fission products during treatment of about 57 metric tons of heavy metal of EBR-II and Fermi-1 blanket fuels. Since declad and clean blanket fuels are packaged and sent to SRS, no additional gaseous fission products are expected to be present in that fuel. However, for conservatism, it was assumed that the gaseous fission products in the blanket fuels would remain within the fuel matrix and would be released to the environment during melt and dilute at SRS. As a result, there would be incurred doses to the public associated with these operations. The duration of the melt and dilute process was estimated to be about 3 years, based on the current design throughput of the melter, and an assumption that the final metallic high-level radioactive waste product from this process would contain about 30 percent depleted uranium in aluminum alloy (WSRC 1999).

Gaseous Emissions:

Based on the assumption that the entire fission gas inventory would remain within the fuel matrix after the decladding and cleaning process, it was assumed the inventory of krypton-85 and tritium would be released during the melt and dilute process. Using the gaseous fission product inventory provided in Appendix D for the EBR-II and Fermi-1 blanket fuels, the potential airborne radiological release quantities were estimated and presented in **Table E–16**. These inventories were then used to calculate the population doses from air emissions.

Table E–16 Annual Radiological Releases During Normal Operations of Melt and Dilute at Building 105-L

Isotope	***Releases[a] to Air (curies)***
H-3	54
Kr-85	399

[a] There are no liquid releases associated with melt and dilute processing at SRS.

Liquid Effluent:

The melt and dilute process would not produce liquid effluent.

Population Impacts:

Estimated annual radiological impacts associated with melt and dilute operations at SRS for the maximally exposed offsite individual and the general population residing within 80-kilometers (50-miles) surrounding Building 105-L, are presented in **Table E–17**. For analysis purposes, the released tritium was assumed to be in oxide form.

Table E–17 Annual Radiological Impacts Due to Normal Operations of Melt and Dilute Process at Building 105-L

Offsite Population		*Maximally Exposed Offsite Individual*	
Dose (person-rem)	*Latent Cancer Fatalities*	*Dose (millirem)*	*Latent Cancer Fatality Risk*
0.0076	3.8×10^{-6}	0.00010	5.0×10^{-11}

Worker Impacts:

Worker impacts were estimated by examining historical doses associated with melt and dilute processing at SRS; these values were based on those presented in the SRS Spent Nuclear Fuel Management Environmental Impact Statement (DOE 1998). The SRS radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. Doses and associated impacts are based on a 3-month processing period. **Table E–18** presents estimated values to both the average worker and entire workforce population.

Table E–18 Annual and Cumulative Worker Radiological Impacts Due to Normal Operations of Melt and Dilute at Building 105-L

Worker Population		*Individual Worker*	
Collective Dose (person-rem per year)	*Latent Cancer Fatalities from 3 years Melt and Dilute Processing*	*Individual Dose (millirem per year)*	*Latent Cancer Fatality Risk from 3 years Melt and Dilute Processing*
50	0.060	500	0.00060

E.4.5 Melt and Dilute at ANL-W (Alternative 6)

In Alternative 6, the blanket and driver fuel elements will need to be prepared at the ANL-W facilities prior to their processing at ANL-W. Preparation of the fuel assemblies at ANL-W for the melt and dilute process requires only that the fuel be cleaned to remove sodium prior to melt and dilute processing; decladding of the blanket and driver fuel is not necessary. This activity would be performed in the argon cell of the Hot Fuel Examination Facility. The following discussion addresses the radiological impacts of normal operations at ANL-W for the preparation and melt and dilute treatment of the blanket and driver fuel assemblies.

Gaseous Emissions:

Fuel preparation would occur at the Hot Fuel Examination Facility. These activities would cause gaseous fission products to be released into the argon cell. As stated earlier in Section E.4.1, krypton-85 and elemental tritium are the most prevalent gaseous radionuclides that would be released to the environment. The released

tritium 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 the krypton-85 and elemental tritium. The maximum released curies of radioactive gaseous emissions occurs when preparation of the blanket fuel and the driver fuel are performed simultaneously. This simultaneous operation was estimated to occur over a 6-year period starting in 2003. Based on a blanket fuel preparation throughput of 10 metric tons of heavy metal and a driver fuel process rate of about 1.7 metric tons of heavy metal annually, about 2,162 curies of elemental tritium and 32,650 curies of krypton-85 would be released to the atmosphere annually (see **Table E-19**). This release rate would last about 2 years, or until all of the EBR-II blanket fuels and the driver fuel assemblies are processed, then afterward the release rate would drop during the processing of the Fermi-1 blanket fuels (The release rate for the processing of 10 metric tons of heavy metal of Fermi blanket fuel is less than 1 curie of elemental tritium and 6.6 curies of krypton-85).

Table E–19 Normal Operation Radiological Emissions for Alternative 6

	Facility	*Maximum Processing Rate (metric tons of heavy metal per year)*	*Duration (years)*	*Annual Release (curies)*	
				Tritium	*Krypton-85*
Driver Fuel	Hot Fuel Examination Facility	1.7	2	2091	32,130
Blanket Fuel	Hot Fuel Examination Facility	10	6	71.2	520
Maximum Annual Release				2162	32,650

Population Impacts:

The doses to the maximally exposed offsite individual and the general public residing within the 80 kilometers (50 miles) surrounding ANL-W are presented in **Table E–20**. As stated in Section E.4.1, the dose resulting from the release of tritium is highly dependent upon its chemical form. The doses presented in Table E–20 result from releases that are assumed to be one percent oxidized tritium, the same assumption used in the analysis of Alternative 1. These impacts are calculated for the preparation and processing of the sodium-bonded blanket and driver assemblies and the total impacts.

Table E–20 Maximum Annual Radiological Impacts Due to Normal Operations Under Alternative 6

	Offsite Population		*Maximally Exposed Offsite Individual*	
	Dose (person-rem)	*Latent Cancer Fatalities*	*Dose (millirem)*	*Latent Cancer Fatality Risk*
Driver Fuel	0.012	6.0×10^{-6}	0.0019	9.5×10^{-10}
Blanket Fuel	0.00028	1.4×10^{-7}	0.000048	2.4×10^{-11}
Total	0.012	6.0×10^{-6}	0.0020	1.0×10^{-9}

E.5 IMPACTS OF EXPOSURES TO HAZARDOUS CHEMICALS ON HUMAN HEALTH

The potential impacts of exposure to hazardous chemicals released to the atmosphere were evaluated for routine operations associated with the alternatives analyzed in this EIS.

The receptors considered in these evaluations are members of the public and noninvolved worker. Impacts of exposures to hazardous chemicals for workers directly involved in the treatment process were not quantitatively evaluated because the use of personal protective equipment and engineering process controls would limit their exposure to levels within applicable Occupational Safety and Health Administration Permissible Exposure Limits or American Conference of Governmental Industrial Hygienists Threshold Limit Values.

As a result of releases from routine operations, receptors are expected to be potentially exposed to concentrations of hazardous chemicals that are below those that could cause acutely toxic health effects. Acutely toxic health effects generally result from short-term exposure to relatively high concentrations of contaminants, such as those that may be encountered during facility accidents. Long-term exposure to relatively lower concentrations of hazardous chemicals can produce adverse chronic health effects that include both carcinogenic and noncarcinogenic effects. The health effect endpoints evaluated in this analysis include excess incidences of latent cancers for carcinogenic chemicals, and a spectrum of chemical-specific noncancer health effects (e.g., headaches, membrane irritation, neurotoxicity, immunotoxicity, liver toxicity, kidney toxicity, developmental toxicity, reproductive toxicity, and genetic toxicity) for noncarcinogens.

Methodology

Annual Airborne concentrations of hazardous chemicals were estimated from the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998).

This EIS estimates the noncancer health risks by comparing annual air concentrations of contaminants to the EPA Reference Concentrations published in the Integrated Risk Information System (IRIS). For each noncarcinogenic chemical, potential health risks are estimated by dividing the estimated airborne concentration by the chemical-specific Reference Concentration value to obtain a noncancer hazard quotient:

Noncancer Hazard Quotient = air concentration/Reference Concentrations

Reference Concentrations are estimates (with an uncertainty spanning perhaps an order of magnitude) of a daily exposure to the human population (including sensitive subgroups) that is likely to be without appreciable risk of harmful effects during a lifetime. Hazard Quotients are calculated for each hazardous chemical to which receptors may be exposed. Hazard Quotients for each chemical are summed to generate a Hazard Index. The Hazard Index is an estimate of the total noncancer toxicity potential from exposure to hazardous chemicals. According to EPA risk assessment guidelines, if the Hazard Index value is less than or equal to 1.0, the exposure is unlikely to produce adverse toxic effects. If the Hazard Index exceeds 1.0, adverse noncancer health effects may result from the exposure.

For carcinogenic chemicals, risk is estimated by the following equation:

$$\text{Risk} = \text{CA} \times \text{URF}$$

where:

Risk = a unitless probability of cancer incidence.
CA = contaminant concentration in air (in micrograms/cubic meters).

URF = cancer inhalation unit risk factor (in units of cancers per micrograms/cubic meters).

Cancer unit risk factors are used in risk assessments to estimate an upper-bound lifetime probability of an individual developing cancer as a result of exposure to a particular level of a potential carcinogen.

Assumptions

The airborne pathway is assumed to be the principal exposure route by which the offsite population maximally exposed individual is exposed to hazardous chemicals released from processing facilities. No synergistic or antagonistic effects are assumed to occur from exposure to the hazardous chemicals. Synergistic effects among released contaminants may result in adverse health effects that are greater than those estimated, whereas antagonistic effects among released chemicals may result in less severe health effects than those estimated.

Analysis

The potential impacts of exposure to hazardous chemicals released to the atmosphere during routine operations of the processing facilities are presented in Chapter 4 for each alternative.

E.6 REFERENCES

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APPENDIX F
EVALUATION OF HUMAN HEALTH EFFECTS FROM FACILITY ACCIDENTS

F.1 INTRODUCTION

This appendix presents the methodology and assumptions used for estimating potential impacts and risks associated with both radiological and toxic chemical releases, due to postulated accidents, at the facilities being considered for the processing of sodium-bonded spent nuclear fuel. Analysis of radiological impacts is presented in section F.2. This is followed by a summary of the risk results for the various alternatives analyzed in detail. Chemical risk methodologies and results are presented in section F.3. Information regarding the impacts of normal operations, along with background information on the health impacts from exposure to ionizing radiation is provided in Appendix E.

F.2 IMPACT OF RADIOLOGICAL ACCIDENTS ON HUMAN HEALTH

This section of Appendix F addresses the radiological impacts associated with accidents at management facilities. Potential accident scenarios have been identified for both the Argonne National Laboratories-West (ANL-W) and Savannah River Site (SRS) facilities proposed for the treatment and management of sodium-bonded spent nuclear fuel.

F.2.1 Overview of Methodology and Basic Assumptions

For the radiological evaluation, the GENII computer program (PNL 1988) has been used to calculate radiation doses to the general population and to selected individuals. Appendix E provides the detailed description of this code, and therefore only the GENII data specific to the accident analysis is presented in this appendix.

The impact of radiation exposure was evaluated for the following segments of the population for each accident scenario:

- **Noninvolved Worker—**An individual (a noninvolved worker) located 100 meters (330 feet) from the radioactive material release point.[1] The dose to the noninvolved worker is calculated for the 50th percentile meteorology only, as specified in DOE-STD-1027-92 (DOE 1992). Noninvolved workers are exposed unprotected to the plume for a limited time (a maximum of 5 minutes), receiving exposure via inhalation, air immersion, and ground surface pathways only.

- **Maximally Exposed Offsite Individual—**A hypothetical individual living at the management site boundary and receiving the maximum exposure. The hypothetical member of the public is located directly downwind of the accident and is exposed to radioactivity via inhalation, ingestion, air immersion, and ground surface pathways. The individual would be exposed to the plume for the entire release duration.

- **Population—**The general public living within an 80-kilometer (50-mile) radius of the facility, residing directly downwind of the accident, and receiving the maximum exposure via inhalation, ingestion, air immersion, and ground surface pathways.

[1] *For elevated release, the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 meters (330 feet) for an elevated release.*

The doses to the maximally exposed offsite individual and the general public are calculated for the 50th and 95th percentile meteorological conditions. Meteorology specific to ANL-W and SRS were used in the evaluation. The site-specific meteorological data was obtained in the form of a joint frequency distribution in terms of percentage of time that the wind blows in specific directions for the given midpoint (or average) wind speed and atmospheric stability. Accident consequences were calculated for both 50th and 95th 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. GENII determines 50th and 95th percentile meteorological conditions using site-specific joint frequency distribution weather data.

The following conditions were used in the calculations:

❒ **Meteorological Data**

- Site-specific joint frequency distribution weather data are used to define 50th and 95th percentile meteorological conditions for each processing technology at management sites.

- If a release occurs through a stack, the release is assumed to occur at an elevated level consistent with the site's effluent emission stack height. The effects of plume rise were not credited in the analysis.

- Mixing layer height is 1,000 meters (3,280 feet). Airborne materials freely diffuse in the atmosphere near the ground level in what is known as the mixing depth. A stable layer exists above the mixing depth and restricts vertical diffusion above 1,000 meters.

- Wet deposition is zero (it is assumed that no rains occur to accelerate deposition and reduce the size of the area affected by the release).

- Dry deposition of the cloud is modeled. During movement of the radioactive plume, a fraction of the radioactive material in the plume is deposited on the ground due to gravitational forces. The quantity of deposited radioactive material is proportional to the particle size and deposition velocities (in meters per second). The deposited material contributes to the exposure from ground surface radiation and ingestion.

❒ **Inhalation Data**

- Breathing rate is 330 cubic centimeters per second (0.7 cubic feet per minute) for the worker and the general public at the site boundary and beyond (maximally exposed individual and population) during the passage of the plume; it is 270 cm^3/sec (0.57 $feet^3$/min) for the general public during the other times.

- Exposure during passage of the entire plume is assessed for the maximally exposed offsite individual and the population. Exposures to the noninvolved worker are to a portion of the plume (i.e., the noninvolved worker is exposed to the plume for a limited time) because the worker is assumed to take emergency action.

- Inhalation exposure factors are based on the International Commission on Radiological Protection, Publication 30 (ICRP 1982).

Exposure time assumptions for maximally exposed individuals, workers, and the general public are provided in **Table F–1** below. Since all accident releases are to the air (either gaseous or suspended particulates), drinking water, aquatic food ingestion, and any other pathways that may involve liquid exposure are not examined. Additional information, common to the analysis of the impacts of normal operations and accidents has previously been presented in Appendix E.

Table F–1 GENII Exposure Parameters to Plumes and Soil Contamination (Postulated Accidents)

Maximally Exposed Offsite Individual			*General Population*		
Inhalation and External Exposure			*Inhalation and External Exposure*		
Exposure Time (hours)	*Breathing Rate (cm^3/sec)*	*Soil Contamination (hours)*	*Exposure Time (hours)*	*Breathing Rate (cm^3/sec)*	*Soil Contamination (hours)*
100 percent of Release Time	330	6,136	100 percent of Release Time	330	6,136

cm^3/sec = cubic centimeter per second
Source: PNL 1988.

Radiological impacts to noninvolved workers from postulated accident scenarios were evaluated at onsite locations where a given incident would cause the highest dose. The noninvolved worker was assumed to have an inhalation exposure time of 5 minutes and an external exposure time to soil contamination of 20 minutes. For a ground-level release accident, a noninvolved worker was assumed to be 100 meters from a given release point; for an elevated release, the worker was situated between 200 and 500 meters, depending on the given site's atmospheric dispersion characteristics. All doses to noninvolved workers include a component associated with the intake of radioactivity into the body, and another component resulting from external exposure to direct radiation.

The radiation dose to individuals and the public resulting from exposure to radioactive releases was calculated using the following potential pathways:

- *Air Immersion*—External direct exposure from immersion in the airborne radioactive material
- *Ground Surface*—External direct exposure from radioactive material deposited on the ground
- *Inhalation*—Internal exposure from inhalation of radioactive aerosols and suspended particles
- *Ingestion*—Internal exposure from ingestion of contaminated terrestrial food and animal products.

The radiation dose is estimated by the GENII computer program in a manner recommended by the International Commission on Radiological Protection in Publications 26 and 30 (ICRP 1977, ICRP 1982). Committed dose equivalents[2] are calculated individually for organs such as the gonads, breast, red bone marrow, lungs, thyroid, and bone surface; calculations are combined for the liver, upper large intestine, lower large intestine, small intestine, and stomach. Weighting factors are used for various body organs to calculate weighted or committed effective dose equivalents from radiation inside the body due to inhalation or ingestion. The committed effective dose equivalent value is the summation of the committed dose equivalent to a specific organ weighted, by the relative risk to that organ compared to an equivalent whole-body exposure. Deep-dose equivalent for the external exposure pathways (immersion in the radioactive material and exposure to the ground contamination) and 50-year committed effective dose equivalent for the internal exposure pathways are calculated. The sum of the deep-dose equivalent for external pathways and committed effective dose equivalent for internal pathways is called the total dose in this EIS.

The exposure from ingestion of contaminated terrestrial food and animal products is calculated on a yearly basis. It is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose should exceed that of the protective action guidelines in a radiological accident event (EPA 1991). No reduction of exposure because of protective actions or evacuation of the public was accounted for in this analysis, however. This conservative approach may result in overestimating health effects within an exposed population, but allows for consistent comparisons between alternatives.

[2]*The definitions of committed dose equivalents, committed effective dose equivalents, and total effective dose equivalents are consistent with those given in 10 Code of Federal Regulations (CFR) Part 835, "Occupational Radiation Protection; Final Rule."*

F.2.2 Selection of Facility Accidents for Detailed Evaluations

The alternatives for the treatment of sodium-bonded spent nuclear fuel assume the use of facilities currently in operation, although modifications to SRS Building 105-L will be necessary before it can be used for the melt and dilute alternative. The selection of accident scenarios is based on those evaluated in the facility safety analysis reports.

Postulated facility accident scenarios were developed based on the review of the analyzed accidents in previous safety analysis, risk assessment, and environmental assessment documents at ANL-W and SRS where the sodium-bonded fuel may be handled or processed. After reviewing a wide range of documents, postulated accident scenarios were developed based on information contained in the following:

- *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995a)

- *Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at ANL-West* (DOE 1996a)

- *Fuel Cycle Facility Final Safety Analysis Report, Revision 4* (ANL 1998a)

- *Safety Analysis Report for the Hot Fuel Examination Facility, Rev 00 DRAFT* (ANL 1998b)

- *Accident Assessments for Idaho National Engineering Laboratory Facilities, DOE/ID* (Slaughterbeek et. al. 1995)

- *Safety Analysis-200 Area, Savannah River Site F-Canyon Operation, F-Canyon SAR Addendum* (WSRC 1994)

- *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement,* (DOE 1998)

Based on this review of analyzed accident scenarios at ANL-W and SRS facilities that deal with sodium-bonded fuel, a spectrum of potential accidents was identified. This process started with systematically identifying initiating events, subsequent accident progressions, and onsite or offsite releases. Then, based on accident initiators, selected accidents were grouped into the following three categories:

- Natural phenomena (e.g., earthquake, tornado),
- External events (e.g., aircraft crash), and
- Process-related events (e.g., explosion, nuclear criticality, fire, spills).

The potential process-related events were further subdivided based on the impact the accident has on the accident release factors. High energy events would be expected to damage some of the confinement barriers provided in the facility design and will result in release factors that approach unity. Medium energy events may reduce the effectiveness of the barriers, but are not expected to defeat them; while low energy events would have almost no impact on the ability of the confinement barriers to perform their function.

A review of the accident scenarios indicated that only severe accident conditions (e.g., accidents involving confinement failure) could result in a significant release of radioactive material to the environment or an increase in radiation levels. These severe accident conditions are associated with beyond-design-basis events, combinations of events for which the facility was not specifically designed. While these events may have consequences larger than those associated with design-basis events, their frequency is expected to be much

lower than the design-basis event frequency. Natural phenomena (e.g., earthquake) and fire accidents creating a direct path for releases to the environment, represented the situation with the most consequences to the public. Some types of accidents, such as procedure violations, spills of small materials containing radioactive particles, and most other types of common human error, occur more frequently than the more severe accidents analyzed. However, these accidents do not involve enough radioactive material or radiation to result in significant release to the environment, although the impact to operational personnel may be as significant as that resulting from beyond-design-basis events. The airborne particles from a process-related accident would normally pass through at least one bank and possibly two to four banks of high-efficiency particulate air filters before entering the environment. Spent nuclear fuel handling operations are performed inside such confinement barriers as hot cells or canyon walls. The hot cells are equipped with safety significant features, such as inert gas atmosphere, pressure control, and heat detection. These features are credited when their operability is not compromised by the sequence of events associated with the accident progression.

While severe accidents (also referred to as beyond-design-basis events) are expected to have the most significant impacts on the population, that is the highest consequences, these accidents may not have as significant a risk impact on all receptors as higher frequency, lower consequence accidents. For this reason, higher frequency accident scenarios were included in the accident analysis. Three categories of accidents were identified, and for each category at least one accident scenario was selected for analysis. The three categories consist of abnormal events (defined as events with a frequency of greater than 1×10^{-3} per year), design-basis events (with frequencies between 1×10^{-3} and 1×10^{-6} per year) and the beyond-design-basis events (frequency less than 1×10^{-6} but limited to those greater than 1×10^{-7} per year).

Based on the review of the existing facility analyses and on guidance provided by the U.S. Department of Energy (DOE) in Section 6.9 of *Recommendations for the Preparation of Environmental Assessments and Environmental Impact Statements* (DOE 1993a), the following types of accidents were selected for each processing technology:

- Explosions
- Nuclear criticality
- Fire
- Earthquake
- Aircraft crash
- Spills/drops

Finally, no specific analyses of the results of terrorist or sabotage acts were considered. This is because the existing security measures in effect at the management sites would essentially preclude any sabotage or terrorist activity. In addition, any acts of terrorism are expected to result in consequences that are bounded by the results of the accident scenarios selected for detailed evaluation.

F.2.2.1 Accident Source Terms and Scenario Description

This section describes the accident scenarios and corresponding source terms developed for ANL-W and SRS. The spectrum of accidents described below were used to determine the incremental consequences (public and worker doses) and risks associated with the treatment of sodium-bonded spent nuclear fuel at each site. These accident scenarios are consistent with those evaluated in either the facility safety analysis report, facility/site environmental reports, or various related DOE safety documents. Secondary accidents were considered when identified in the safety documents. The selected documents were identified and referenced in each of the accident scenarios described. When information was required to further clarify the accident condition, update some of the parameters, and facilitate the evaluation process, additional assumptions were made. Sometimes it was necessary to use different assumptions than those that were used in the referenced report, which are also identified. For example, the material at risk during an earthquake can be different for the treatment in this EIS than those considered in the facility safety analysis report. This change in assumption is necessary because the

evaluations in this EIS focus only on the risk resulting from the implementation of alternatives (an incremental risk), and therefore address only the risk associated with the treatment of the sodium-bonded fuel. Cumulative risks can be determined by adding the incremental risks to the existing risks.

F.2.2.1.1 Source Terms

The source term (or building source term) is the amount of respirable radioactive material that is released to the air, in terms of curies or grams, assuming the occurrence of a postulated accident. The airborne source term is typically estimated by the following five-component linear equation:

$$\text{Source term} = \text{MAR} \times \text{DR} \times \text{ARF} \times \text{RF} \times \text{LPF}$$

where:

MAR	=	Material-at-Risk (grams or curies)
DR	=	Damage Ratio
ARF	=	Airborne Release Fraction (or Airborne Release Rate for continuous release)
RF	=	Respirable Fraction
LPF	=	Leak Path Factor

- **Material at Risk—**The material at risk is the amount of the radionuclides (in curies of activity or grams for each radionuclide) available for release when acted upon by a given physical stress (i.e., an accident). The material at risk is specific to a given process in the facility of interest. It is not necessarily the total quantity of material present, but is that amount of material in the scenario of interest postulated to be available for release.

- **Damage Ratio—**This is the fraction of material exposed to the effects of the energy/force/stress generated by the postulated event. For the accident scenarios discussed in this document, the value of the damage ratio varies from 1×10^{-4} to 1.0.

- **Airborne Release Fraction—**This is the fraction of material that becomes airborne due to the accident. In this analysis, airborne release fraction values from the DOE Handbook on airborne release fraction are used (DOE 1994b).

- **Respirable Fraction—**This is the fraction of the material, with particle size of 10-micrometers (microns) aerodynamic equivalent diameter or less, that could be retained in the respiratory system following inhalation. The respirable fraction values are also taken from the DOE Handbook on airborne release fractions (DOE 1994b).

- **Leak Path Factor—**The leak path factor accounts for the action of removal mechanisms (e.g., containment systems, filtration, deposition) to reduce the amount of airborne radioactivity that is ultimately released to occupied spaces in the facility or the environment. A leak path factor of 1 (i.e., no reduction) is assigned in accident scenarios involving a major failure of confinement barriers.

F.2.2.1.2 Accident Scenarios Description and Source Terms at ANL-W

- **Description of Accident Scenarios for Electrometallurgical Treatment Process—**The electrometallurgical treatment process would occur at the Fuel Conditioning Facility and the Hot Fuel Examination Facility at the ANL-W site. This process is detailed in Appendix C. The accident scenarios, identified in **Table F–2** and defined in the following paragraphs, are applicable to the electrometallurgical treatment process as proposed at ANL-W. This section also provides information addressing the material at risk and the various release fractions used to determine the source term for each accident selected for analysis.

Table F–2 Selected Accident Scenarios for the Electrometallurgical Treatment Process at ANL-W

Scenario	*Frequency (per year)*
Process Related Spills/Drops a. Salt Powder Spill b. Cask Drop c. Salt Canister Transfer Accident, Canister Breach	 1×10^{-2} 1×10^{-2} 1×10^{-7}
Single Container Solid Transuranic Waste Fire	1×10^{-3}
Explosion [a]	N/A
Earthquake (DBE)	2×10^{-4} [b] / 0.008 [c]
Aircraft Crash	6×10^{-7} to 1×10^{-8}
Nuclear Criticality	less than 10^{-7}
Earthquake (BDBE)	1×10^{-5}

DBE = design- (evaluation) basis earthquake, BDBE = beyond-design-basis earthquake.
[a] N/A - The explosion scenario is not applicable to this process.
[b] Design-basis earthquake for the Fuel Conditioning Facility.
[c] Design-basis earthquake for the Hot Fuel Examination Facility.

Each accident scenario description sets the condition of the accident and provides a summary of material involved. As stated earlier, some of these accident scenarios are generic, but their applications are consistent with those evaluated in various ANL-W environmental and safety analyses. These accidents include process-specific as well as generic and process-independent accidents. Tables F–3 through F–8 provide a summary of the accidents analyzed, the material at risk, and the release factors based on the sodium-bonded spent nuclear fuel type that is expected to produce the most significant consequences, typically either Experimental Breeder Reactor-II blanket or driver fuel, for each postulated accident.

- *Operational accident causing salt powder spill in Hot Fuel Examination Facility Main Cell*—Solidified electrorefiner salt is sent from the Fuel Conditioning Facility to the Hot Fuel Examination Facility for processing into the final ceramic waste form. It is brought into the Hot Fuel Examination Facility in the solid form and ground. The grinder is located in the Hot Fuel Examination Facility Main Cell on a raised floor. In this accident scenario, it was assumed that during a transfer operation, the contents of a ground salt container is spilled and the powder spills into the pit beneath the floor. A portion of the salt powder becomes airborne and is carried through the ventilation system to the high-efficiency particulate air filters, and released through the building stack. The release is assumed to occur over a 1-hour period. The frequency of this accident was set at 1×10^{-2} per year, based on the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b).

  The salt is assumed to come from the treatment of 4.45 metric tons of heavy metal of EBR-II blanket elements or 1.1 metric tons of heavy metal of EBR-II driver elements, the point at which the salt is conservatively assumed to have been replaced during processing (Goff et al 1999b). Based on the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b), the material at risk was assumed to be 100 kilograms ground salt containing the radionuclide concentrations as shown in **Table F–3**. Radionuclide distributions were developed for both EBR-II driver and blanket fuels. The radionuclide distribution for driver fuel is based on an average plutonium concentration in electrorefiner salt of 1.76 percent by weight. The radionuclide distribution for blanket fuel is based on an average plutonium concentration in electrorefiner salt of 6.76 percent by weight (Goff et al 1999). Portions of the spilled salt will become airborne. The maximum measured value for the three meter free-fall of dry cohesionless particles, with a mass median diameter of 1 to 2 microns, result in an airborne release fraction of 0.002 and an respirable fraction of 0.3 (DOE 1994b). The median particle size of the salt after grinding is approximately 200 microns with only about 1 percent being of a diameter of less than 20 microns (ANL 1999). The analysis therefore conservatively assumed that about

1 percent of the ground salt would have characteristics capable of resulting in the airborne release fraction and respirable fractions identified above, resulting in a damage ratio of 0.01. The ventilation system and high-efficiency particulate air filters are assumed to function normally. The ventilation system consists of a two-stage high-efficiency particulate air filtration system or equivalent, with a first stage high-efficiency particulate air filter efficiency of 99.9 percent, and a second stage efficiency of 99 percent. The leak path factor through the high-efficiency particulate air filters is therefore 1×10^{-5}.

Table F–3 Material at Risk and Release Fraction Values for the Salt Power Spill Accident at ANL-W

Material at Risk[a]							*Source Terms (curies)*	
Isotope	***Blanket (curies)***	***Driver (curies)***	***DR***	***ARF***	***RF***	***LPF***	***Blanket***	***Driver***
Sr-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	1.0E-05	2.96E-08	2.10E-06
Y-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	1.0E-05	2.96E-08	2.10E-06
I-129	8.79E-04	1.31E-02	0.01	2.0E-03	3.0E-01	1.0E-05	5.27E-14	7.86E-13
Cs-134	8.18E+00	3.13E+02	0.01	2.0E-03	3.0E-01	1.0E-05	4.91E-10	1.88E-08
Cs-137	1.06E+03	3.92E+04	0.01	2.0E-03	3.0E-01	1.0E-05	6.36E-08	2.35E-06
Ba-137M	1.00E+03	3.71E+04	0.01	2.0E-03	3.0E-01	1.0E-05	6.00E-08	2.23E-06
Ce-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	1.0E-05	2.30E-09	3.16E-08
Pr-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	1.0E-05	2.30E-09	3.16E-08
Pm-147	2.48E+02	1.47E+04	0.01	2.0E-03	3.0E-01	1.0E-05	1.49E-08	8.82E-07
Sm-151	6.10E+01	9.48E+02	0.01	2.0E-03	3.0E-01	1.0E-05	3.66E-09	5.69E-08
Eu-154	4.48E+00	1.01E+02	0.01	2.0E-03	3.0E-01	1.0E-05	2.69E-10	6.06E-09
Eu-155	2.94E+01	6.77E+02	0.01	2.0E-03	3.0E-01	1.0E-05	1.76E-09	4.06E-08
Th-228	9.46E-05	9.13E-03	0.01	2.0E-03	3.0E-01	1.0E-05	5.68E-15	5.48E-13
Np-237	7.94E-05	5.13E-02	0.01	2.0E-03	3.0E-01	1.0E-05	4.76E-15	3.08E-12
Pu-238	5.15E+00	6.68E+01	0.01	2.0E-03	3.0E-01	1.0E-05	3.09E-10	4.01E-09
Pu-239	4.13E+02	1.08E+02	0.01	2.0E-03	3.0E-01	1.0E-05	2.48E-08	6.48E-09
Pu-240	2.84E+01	3.67E+00	0.01	2.0E-03	3.0E-01	1.0E-05	1.70E-09	2.20E-10
Pu-241	1.15E+02	8.93E+00	0.01	2.0E-03	3.0E-01	1.0E-05	6.90E-09	5.36E-10
Am-241	9.95E+00	6.94E-02	0.01	2.0E-03	3.0E-01	1.0E-05	5.97E-10	4.16E-12
Am-242M	1.03E-01	5.88E-05	0.01	2.0E-03	3.0E-01	1.0E-05	6.18E-12	3.53E-15

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

[a] Radionuclide Inventory from Appendix D.

- *Cask drop and gaseous fission product release*—Spent nuclear fuel casks will be handled frequently when the sodium-bonded fuel is processed. (Spent nuclear fuel handling at the ANL-W site is not limited to that associated with the treatment of the sodium-bonded spent nuclear fuel. The accident discussed here is intended to address only that portion of the handling activity that can be directly attributed to the treatment of sodium-bonded spent nuclear fuel.) Spent nuclear fuel stored in the Radioactive Scrap and Waste Facility will be transferred to the Fuel Conditioning Facility for processing, and spent nuclear fuel will be received from off site at the Hot Fuel Examination Facility and transferred between the Hot Fuel Examination Facility and the Fuel Conditioning Facility. The HFEF-5 cask would be used to move EBR-II driver and blanket fuels from the Radioactive Scrap and Waste Facility, to the Fuel Conditioning Facility. The postulated accident is described in the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b). The accident

involves a dropped cask during unloading, resulting in seal failure and fuel cladding failure sufficient to release gaseous and volatile fission products to the atmosphere. The drop could be initiated by failure of lifting equipment, failure of slings, hooks, or cables, or human error by the lifting equipment operator. The cask drop is conservatively assumed to result in an unfiltered release of gaseous and volatile fission products. The release is assumed to be a puff release at ground level. Dropping of casks, while rare, is nevertheless categorized as being anticipated since such events have happened in the past and may be expected to occur over the lifetime of the facility. The frequency of cask dropping is assumed to be 1×10^{-2} per year, consistent with that used in the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b).

The HFEF-5 cask can contain two EBR-II driver subassemblies. It is conservatively assumed that the equivalent of one subassembly (61 elements) fails in the accident. The material at risk, as shown in **Table F–4**, is the equivalent of one EBR-II driver or one EBR-II blanket subassembly. The damage ratio for the failed elements is assumed to be 1, since all gaseous and volatile fission products conservatively could be released to the cask following cladding failure. The airborne release fraction/respirable fraction for gases is assumed to be 1, and 1×10^{-7} for cesium from the dislodgement of surface contamination (DOE 1995a). The accident is assumed to occur outdoors, so a leak path factor of 1 is assumed.

Table F–4 Material at Risk and Release Fraction Values for the Cask Drop Accident at ANL-W

Material at Risk[a]							*Source Terms (curies)*	
Isotope	***Blanket (curies)***	***Driver (curies)***	***DR***	***ARF***	***RF***	***LPF***	***Blanket***	***Driver***
H-3[b]	3.35E-01	5.17E+00	1	1	1	1	3.35E-01	5.17E+00
Kr-85	2.44E+00	7.94E+01	1	1	1	1	2.44E+00	7.94E+01
Cs-134	6.30E-01	7.39E+00	1	1.0E-07	1	1	6.30E-08	7.39E-07
Cs-137	8.13E+01	9.28E+02	1	1.0E-07	1	1	8.13E-06	9.28E-05

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF=leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

[a] Data for 1 assembly based on Appendix D data for curie content.

[b] Assumes 1 percent becomes oxidized.

- *Salt spill during transfer from Fuel Conditioning Facility to Hot Fuel Examination Facility*—Solidified electrorefiner salt is sent from the Fuel Conditioning Facility to the Hot Fuel Examination Facility for processing into the final ceramic waste form. It is transferred in the form of large chunks within the HFEF-5 cask. Transfer is via forklift or truck. In this scenario, a severe vehicle accident occurs resulting in breach of the inner and outer salt container. The accident could be caused by operator error or equipment failure. The accident is considered beyond-design-basis due to the durability of the shielded HFEF-5 canister. There will be over 200 transfers of salt from the Fuel Conditioning Facility to the Hot Fuel Examination Facility. A probability of 1×10^{-7} is assumed. The release occurs at ground level with a duration of 1 hour.

  Table F–5 provides the isotopic material at risk for a total material at risk of 20 kilograms of salt per transfer based on discussions with ANL-W personnel. The salt is in the forms of chunks (i.e., ice cube size) and is not combustible. No significant release is assumed from the large pieces. Some of the salt pieces would experience brittle fracture and release of particulate. A brittle fracture particulate fraction for solidified salt is 1×10^{-4} for particles less than 10 microns in diameter (ANL 1998b), therefore, a damage ratio of 1×10^{-4} is assumed. Conservatively, the same airborne release fraction/respirable fraction values as were used for the spill of the salt powder in the Hot Fuel Examination Facility Main Cell were used, that is airborne release fraction for powder is 0.002 and the respirable fraction is 0.3 (DOE 1994b). The accident occurs outdoors, therefore the leak path factor is 1.0.

Table F–5 Material at Risk and Release Fraction Values for the Salt Transfer Accident at ANL-W

Material at Risk[a]							*Source Terms (curies)*	
Isotopes	*Blanket (curies)*	*Driver (curies)*	*DR*	*ARF*	*RF*	*LPF*	*Blanket*	*Driver*
Sr-90	9.85E+01	7.00E+03	1.0E-04	2.0E-03	3.0E-01	1	5.91E-06	4.20E-04
Y-90	9.85E+01	7.00E+03	1.0E-04	2.0E-03	3.0E-01	1	5.91E-06	4.20E-04
I-129	1.76E-04	2.61E-03	1.0E-04	2.0E-03	3.0E-01	1	1.06E-11	1.57E-10
Cs-134	1.64E+00	6.25E+01	1.0E-04	2.0E-03	3.0E-01	1	9.84E-08	3.75E-06
Cs-137	2.11E+02	7.85E+03	1.0E-04	2.0E-03	3.0E-01	1	1.27E-05	4.71E-04
Ba-137M	2.00E+02	7.42E+03	1.0E-04	2.0E-03	3.0E-01	1	1.20E-05	4.45E-04
Ce-144	7.65E+00	1.05E+02	1.0E-04	2.0E-03	3.0E-01	1	4.59E-07	6.30E-06
Pr-144	7.65E+00	1.05E+02	1.0E-04	2.0E-03	3.0E-01	1	4.59E-07	6.30E-06
Pm-147	4.97E+01	2.93E+03	1.0E-04	2.0E-03	3.0E-01	1	2.98E-06	1.76E-04
Sm-151	1.22E+01	1.90E+02	1.0E-04	2.0E-03	3.0E-01	1	7.32E-07	1.14E-05
Eu-154	8.96E-01	2.01E+01	1.0E-04	2.0E-03	3.0E-01	1	5.38E-08	1.21E-06
Eu-155	5.87E+00	1.35E+02	1.0E-04	2.0E-03	3.0E-01	1	3.52E-07	8.10E-06
Th-228	1.89E-05	1.83E-03	1.0E-04	2.0E-03	3.0E-01	1	1.13E-12	1.10E-10
Np-237	1.59E-05	1.03E-02	1.0E-04	2.0E-03	3.0E-01	1	9.54E-13	6.18E-10
Pu-238	1.03E+00	1.34E+01	1.0E-04	2.0E-03	3.0E-01	1	6.18E-08	8.04E-07
Pu-239	8.25E+01	2.16E+01	1.0E-04	2.0E-03	3.0E-01	1	4.95E-06	1.30E-06
Pu-240	5.68E+00	7.33E-01	1.0E-04	2.0E-03	3.0E-01	1	3.41E-07	4.40E-08
Pu-241	2.30E+01	1.79E+00	1.0E-04	2.0E-03	3.0E-01	1	1.38E-06	1.07E-07
Am-241	1.99E+00	1.39E-02	1.0E-04	2.0E-03	3.0E-01	1	1.19E-07	8.34E-10
Am-242M	2.06E-02	1.18E-05	1.0E-04	2.0E-03	3.0E-01	1	1.24E-09	7.08E-13

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF=leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
[a] The material at risk is the isotope in 20 kilograms of salt, which is 20 percent of those given in Table F-3.

- *Solid transuranic waste fire*—Transuranic waste is generated as a result of treatment operations, as well as for other operations at ANL-W. These wastes are placed in containers and temporarily stored (staged) at ANL-W pending shipment to the Radioactive Waste Management Complex. A fire is postulated to occur in a 1.2 × 1.2 × 2.4-meter (4 × 4 × 8-foot) transuranic waste box due to spontaneous combustion, pyrophoric material, vehicle accident, electrical failure, or poor housekeeping. The fire consumes the contents of one box of staged transuranic waste. The accident is assumed to occur outdoors during handling. The release occurs at ground level over 1 hour. An accident frequency in the range of Unlikely (1×10^{-4} to 1×10^{-2}) is documented in the Final Safety Analysis Report for the Fuel Conditioning Facility (ANL 1998a). Here, the accident was assumed to have a frequency of 1×10^{-3} per year.

  The material at risk, as shown in **Table F–6**, was assumed to be one box of transuranic waste. The waste boxes are loaded with 1/20th of 0.34 curies of alpha activity, as described in the Fuel Conditioning Facility Final Safety Analysis Report (ANL 1998a). The material at risk is 0.017 curies of transuranic nuclides, with the nuclide distribution associated with the generic contents of a transuranic waste container. The damage ratio is assumed to be 1.0, since all waste in the container is assumed to be involved in the fire. The DOE Handbook *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE 1994b) recommends an airborne release fraction of 5×10^{-4} and a respirable fraction of 1.0 for burning of surface contaminated wastes. The leak path factor is assumed to be 1.0. No credit is taken for building confinement.

Table F–6 Material at Risk and Release Fraction Values for the Transuranic Waste Fire Accident at ANL-W

Material at Risk [a]						
Isotope	***curies***	***DR***	***ARF***	***RF***	***LPF***	***Source Term (curies)***
Pu-238	1.53E-04	1	5.00E-04	1.00E+01	1	7.67E-08
Pu-239	1.23E-02	1	5.00E-04	1.00E+01	1	6.15E-06
Pu-240	8.46E-04	1	5.00E-04	1.00E+01	1	4.23E-07
Pu-241	3.43E-03	1	5.00E-04	1.00E+01	1	1.72E-06
Am-241	2.66E-04	1	5.00E-04	1.00E+01	1	1.33E-07

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF=leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
[a] The MAR is for a generic waste package, not for any specific spent nuclear fuel.

- *Design-Basis Seismic Event - Multi-facility Effects*—In the Fuel Conditioning Facility, the argon cell contains the equipment for processing sodium-bonded spent nuclear fuel into salt and metal waste forms and uranium metal product. All operations involving bare fuels are conducted in the argon cell because the inert atmosphere precludes pyrophoric metal fire. Fire cannot occur unless sufficient oxygen enters the cell through a cell breach. The walls, ceiling, and floor of the argon cell are constructed from reinforced concrete with thicknesses ranging from 1.2 to 1.5 meters (4 to 5 feet). It also has a gas-tight steel lining. It is assumed that the accident occurs during electrometallurgical treatment operations. Chopped fuel, electrorefiner salts, cathodes, and anodes are all present in the argon cell. At the Fuel Conditioning Facility, a seismic event results in cell breach and inlet of air to the cell. The air in the cell causes pyrophoric metals to ignite and burn. The electrorefiners are seismically qualified, and no spill of molten salt is postulated. The Fuel Conditioning Facility Safety Exhaust System is seismically qualified, and is assumed to function as designed, filtering the cell atmosphere prior to release through the Fuel Conditioning Facility stack.

The Final Safety Analysis Report for the Fuel Conditioning Facility (ANL 1998a) identifies the seismic design goal for the facility to be the ability to withstand a 0.21g design-basis seismic event. This event is identified as having a return frequency of 2×10^{-4} per year. The safety exhaust system will remain operational, although breaches may occur in the argon-cell boundary, after a design-basis earthquake. The Safety Exhaust Building, which includes the high-efficiency particulate air filters, is designed to withstand an earthquake of 0.24 g. In the Hot Fuel Examination Facility, grinding of salt into powder is assumed to be occurring in the main cell. The grinder is located in the Hot Fuel Examination Facility main cell on a raised floor consisting of steel plates resting on supports. Underneath the floor is a 2.4-meter- (8-foot-) deep pit that houses the ventilation ductwork and high-efficiency particulate air filters. At the Hot Fuel Examination Facility, a seismic event causes the vessel containing ground salt to topple and the powder to spill out. Since the ventilation system is not seismically qualified, it is assumed to fail and result in an unfiltered release. It is also assumed that the seismic event would cause a loss of electrical power, which would also result in failure of the ventilation system. The main cell breaches at piping or ventilation penetrations, providing a release path for the suspended powder. The releases occur over a 1-hour period, and are modeled as a ground-level release. The Hot Fuel Examination Facility has been analyzed for a 0.14 g design-basis seismic event, an event with a return frequency of 0.001 per year and a performance goal of 1×10^{-4} per year. No records exist on the original equipment design. However, all major systems are known to have survived the 0.03 Borah earthquake in 1983, an event with a return frequency of 0.008 per year. While it is expected that the equipment would survive the 0.14 g earthquake, the 0.008 per year (ANL 1998b) seismic event frequency has been conservatively used to represent the upper bound of the design-basis seismic event which would result in a salt powder spill and the possible failure of the ventilation system. This frequency is nearly two orders of magnitude higher than the 0.21 g earthquake that could impact both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Therefore, 0.008 per year is used for the design-basis seismic accident frequency.

In the Fuel Conditioning Facility, the material at risk is chopped fuel and cathodes in the argon cell at the time of the accident. **Table F–7** provides material at risk values for the isotopes of concern. The bounding inventory is 20 kilograms (37 pounds) of chopped fuel and 25 kilograms (44 pounds) in two solid electrodes. The solid

cathodes contain 17 kilograms uranium. Uranium is considered a toxic chemical in the consequence assessment. The total uranium material at risk is 19 kilograms. During the postulated event, 100 kilograms (220 pounds) of solidified salt powder with the same concentration of radionuclides as described above for the powder is spilled in the Hot Fuel Examination Facility main cell. For the metal fire in the Fuel Conditioning Facility argon cell, the damage ratio is assumed to be 1.0, since all materials in the material at risk are released to the cells in the accident. As with the previously discussed salt powder spill, only a fraction of the material is small enough to be capable of resuspended, and the damage ratio is 0.01. For the Fuel Conditioning Facility, the airborne release fraction/respirable fraction is 1.0 for krypton-85, 0.00025 for cesium and 2.5×10^{-6} for strontium, uranium, and transuranic waste nuclides (DOE 1995a). For the Hot Fuel Examination Facility powder spill within the cell, an airborne release fraction of 0.002 and an respirable fraction of 0.3 are assumed (DOE 1994b). These are the same values as used for the salt powder spill accident described above. For the Fuel Conditioning Facility, the safety exhaust system remains functional, and the release is filtered through high-efficiency particulate air filters. A leak path factor of 1×10^{-5} is assumed for all particulates. The Hot Fuel Examination Facility leak path for the release is through three enclosures before reaching the outside: main cell, ducts and pipes, and the building. Consistent with the facility safety analysis report assumption, a leak path factor of 0.5 is assigned to each enclosure for plate out and settling of the airborne powder. Therefore, the total leak path factor is $0.5 \times 0.5 \times 0.5 = 0.125$.

Table F–7 Material at Risk and Release Fraction Values for the Design Basis Seismic Event at ANL-W

	Material at Risk							*Source Term (curies)*	
Accident	***Isotope***	***Blanket (curies)***	***Driver (curies)***	***DR***	***ARF***	***RF***	***LPF***	***Blanket***	***Driver***
Seismic event and salt powder spill at the Hot Fuel Examination Facility	Sr-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	0.125	3.70E-04	2.63E-02
	Y-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	0.125	3.70E-04	2.63E-02
	I-129	8.79E-04	1.31E-02	0.01	2.0E-03	3.0E-01	0.125	6.59E-10	9.83E-09
	Cs-134	8.18E+00	3.13E+02	0.01	2.0E-03	3.0E-01	0.125	6.14E-06	2.35E-04
	Cs-137	1.06E+03	3.92E+04	0.01	2.0E-03	3.0E-01	0.125	7.95E-04	2.94E-02
	Ba-137M	1.00E+03	3.71E+04	0.01	2.0E-03	3.0E-01	0.125	7.50E-04	2.78E-02
	Ce-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	0.125	2.87E-05	3.95E-04
	Pr-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	0.125	2.87E-05	3.95E-04
	Pm-147	2.48E+02	1.47E+04	0.01	2.0E-03	3.0E-01	0.125	1.86E-04	1.10E-02
	Sm-151	6.10E+01	9.48E+02	0.01	2.0E-03	3.0E-01	0.125	4.58E-05	7.11E-04
	Eu-154	4.48E+00	1.01E+02	0.01	2.0E-03	3.0E-01	0.125	3.36E-06	7.58E-05
	Eu-155	2.94E+01	6.77E+02	0.01	2.0E-03	3.0E-01	0.125	2.21E-05	5.08E-04
	Th-228	9.46E-05	9.13E-03	0.01	2.0E-03	3.0E-01	0.125	7.10E-11	6.85E-09
	Np-237	7.94E-05	5.13E-02	0.01	2.0E-03	3.0E-01	0.125	5.96E-11	3.85E-08
	Pu-238	5.15E+00	6.68E+01	0.01	2.0E-03	3.0E-01	0.125	3.86E-06	5.01E-05
	Pu-239	4.13E+02	1.08E+02	0.01	2.0E-03	3.0E-01	0.125	3.10E-04	8.10E-05
	Pu-240	2.84E+01	3.67E+00	0.01	2.0E-03	3.0E-01	0.125	2.13E-05	2.75E-06
	Pu-241	1.15E+02	8.93E+00	0.01	2.0E-03	3.0E-01	0.125	8.63E-05	6.70E-06
	Am-241	9.95E+00	6.94E-02	0.01	2.0E-03	3.0E-01	0.125	7.46E-06	5.21E-08
	Am-242M	1.03E-01	5.88E-05	0.01	2.0E-03	3.0E-01	0.125	7.73E-08	4.41E-11
Seismic event and metal fire in the Fuel Conditioning Facility argon cell	H-3	1.42E-01	2.46E+01	1	1	1	1	1.42E-01	2.46E+01
	C-14	1.19E-03	3.98E+03	1	2.5E-06	1	1E-05	2.99E-14	9.95E-08
	Fe-55	1.80E+00	9.74E+01	1	2.5E-06	1	1E-05	4.51E-11	2.44E-09
	Ni-63	6.12E-02	4.58E+00	1	2.5E-06	1	1E-05	1.53E-12	1.15E-10
	Kr-85	1.04E+00	3.78E+02	1	1	1	1	1.04E+00	3.78E+02
	Sr-90	1.61E+01	3.94E+03	1	2.5E-06	1	1E-05	4.04E-10	9.85E-08
	Y-90	1.61E+01	3.94E+03	1	2.5E-06	1	1E-05	4.04E-10	9.85E-08
	Ru-106	2.70E+00	3.02E+01	1	2.5E-04	1	1E-05	6.75E-09	7.55E-08
	Rh-106	2.70E+00	3.02E+01	1	2.5E-06	1	1E-05	6.75E-11	7.55E-10

Accident	Material at Risk			DR	ARF	RF	LPF	Source Term (curies)	
	Isotope	Blanket (curies)	Driver (curies)					Blanket	Driver
	Cd-113M	1.42E-02	9.28E-01	1	2.5E-06	1	1E-05	3.56E-13	2.32E-11
	Sb-125	4.62E-01	5.92E+01	1	2.5E-06	1	1E-05	1.16E-11	1.48E-09
	Te-125M	1.90E-01	2.46E+01	1	2.5E-06	1	1E-05	4.76E-12	6.15E-10
	I-129	2.88E-05	1.47E-03	1	1	1	1	2.88E-05	1.47E-03
	Cs-134	2.68E-01	3.52E+01	1	2.5E-04	1	1E-05	6.70E-10	8.80E-08
	Cs-137	3.46E+01	4.42E+03	1	2.5E-04	1	1E-05	8.65E-08	1.11E-05
	Ba-137M	3.28E+01	4.18E+03	1	2.5E-06	1	1E-05	8.20E-10	1.05E-07
	Ce-144	1.25E+00	5.92E+01	1	2.5E-06	1	1E-05	3.14E-11	1.48E-09
	Pr-144	1.25E+00	5.92E+01	1	2.5E-6	1	1E-05	3.14E-11	1.48E-09
	Pm-147	8.14E+00	1.65E+03	1	2.5E-06	1	1E-05	2.04E-10	4.13E-08
	Sm-151	2.00E+00	1.07E+02	1	2.5E-06	1	1E-05	5.00E-11	2.67E-09
	Eu-154	1.47E-01	1.13E+01	1	2.5E-06	1	1E-05	3.67E-12	2.84E-10
	Eu-155	9.62E-01	7.62E+01	1	2.5E-06	1	1E-05	2.41E-11	1.91E-09
	Th-228	3.10E-06	1.03E-03	1	2.5E-06	1	1E-05	7.75E-17	2.57E-14
	U-234	2.66E-05	8.08E-01	1	2.5E-06	1	1E-05	6.65E-16	2.02E-11
	U-235	7.54E-05	2.62E-02	1	2.5E-06	1	1E-05	1.89E-15	6.55E-13
	U-236	8.48E-05	2.42E-02	1	2.5E-06	1	1E-05	2.12E-15	6.05E-13
	U-238	6.54E-03	2.22E-03	1	2.5E-06	1	1E-05	1.64E-13	5.55E-14
	Np-237	2.60E-06	5.78E-03	1	2.5E-06	1	1E-05	6.50E-17	1.45E-13
	Pu-238	1.88E-01	3.32E+00	1	2.5E-06	1	1E-05	4.70E-12	8.30E-11
	Pu-239	1.51E+01	5.38E+00	1	2.5E-06	1	1E-05	3.77E-10	1.35E-10
	Pu-240	1.04E+00	1.82E-01	1	2.5E-06	1	1E-05	2.59E-11	4.56E-12
	Pu-241	4.20E+00	4.44E-01	1	2.5E-06	1	1E-05	1.05E-10	1.11E-11
	Am-241	3.26E-01	7.82E-03	1	2.5E-06	1	1E-05	8.15E-12	1.96E-13
	Am-242M	3.38E-03	6.62E-06	1	2.5E-06	1	1E-05	8.45E-14	1.66E-16

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

- *Aircraft crash*—The potential for an aircraft crash was evaluated. The methodology for evaluating the likelihood of an aircraft crash is documented in DOE Standard *Accident Analysis for Aircraft Crash into Hazardous Facilities* (DOE 1996c). At INEEL, the probability of a small and large aircraft crash is 9×10^{-5} and 4×10^{-7} crashes per square mile per year, respectively. Using guidance in this DOE standard, the effective area of the Fuel Conditioning Facility was calculated accounting for aircraft wing span and potential skid distance. The effective area of the Fuel Conditioning Facility is about 0.03 square miles for a large aircraft, and 0.007 square miles for a small aircraft. The effective area of the Fuel Conditioning Facility is conservative because the area of the air cell and argon cell, where the hazardous material are contained, are smaller than the total area of the building. Multiplying effective area by INEEL-specific crash rates, gives an estimated probability of a crash into the Fuel Conditioning Facility of 1×10^{-8} for large aircraft and 6×10^{-7} for small aircraft. Comparable probabilities are applicable to the Hot Fuel Examination Facility. A large aircraft crash is not reasonably foreseeable, and given the 1.2 to 1.5-meter- (4 to 5-foot-) thick walls of the cells and the "buffer" provided by the building exterior walls, crash of a small aircraft is unlikely to result in any damage to the cells. Damage from the more probable seismic events analyzed are considered to bound the damage that could result from a small aircraft crash. Also, seismic events affect more than one facility, where an aircraft crash could only affect one facility. Therefore, an aircraft crash is not analyzed separately.

- *Nuclear Criticality*—The potential for a nuclear criticality was considered in the accident analysis. Nuclear criticality has been evaluated in the safety analyses documented for the ANL-W facilities, as required by DOE. The existing safety analyses conclude that nuclear criticality is beyond the design-basis of the facilities

proposed for the electrometallurgical treatment alternative, and therefore has a probability of less than 1×10^{-6} per year. This conclusion is based on a lack of nuclear moderator materials, equipment design, and inventory controls, as well as numerous other administrative controls and operating procedures. The intent of the process is to dilute, rather than concentrate, fissile materials. Fuel storage racks and processing equipment are designed to maintain their safety function during the design-basis seismic event. Even in a beyond-design-basis earthquake (maximum frequency 1×10^{-5} per year), nuclear materials would have to come together in an ideal critical array in order for criticality to be possible. For example, it would require more than the equivalent of 10 EBR-II driver assemblies (610 individual elements) in an ideal geometric configuration to create a potential criticality hazard. The conditional probability of this configuration, given a beyond-design-basis seismic event, is estimated to be no greater than 1×10^{-2}. Therefore, criticality is not considered to be reasonably foreseeable, and is not analyzed quantitatively.

- *Beyond-Design-Basis Seismic Event*—The scenario is similar to the design-basis seismic event, except that the Safety Exhaust System is not assumed to function at the Fuel Conditioning Facility, and an electrorefiner is assumed to spill its molten salt. Also, since fuel is stored in both the Fuel Conditioning Facility and the Hot Fuel Examination Facility, the equivalent of twelve assemblies of EBR-II driver fuel are assumed to experience cladding failure and release of gaseous and volatile fission products. All releases are modeled as ground-level releases. The Fuel Conditioning Facility natural phenomena hazard performance goal is a frequency of 10^{-5} (DOE 1994a). (The Hot Fuel Examination Facility goal is 1×10^{-4}.) The performance goal can be interpreted as the frequency level at which facility damage will initiate. The Fuel Conditioning Facility and Safety Exhaust System are not expected to suffer damage from seismic events with frequencies higher than this. The Fuel Conditioning Facility horizontal acceleration seismic design-basis is 0.21 g, and the newer safety equipment building is designed for a 0.24 g horizontal acceleration. An 0.24 g peak acceleration corresponds to an earthquake frequency at ANL-W of approximately 1×10^{-4} per year (WCFS 1996). An earthquake with a peak ground acceleration of 0.24 g may damage the Fuel Conditioning Facility structure. Therefore, the upper bound for the beyond-design-basis seismic event frequency has been assumed to correspond to the frequency of the performance goal, 10^{-5} per year.

The material at risk, provided in **Table F–8**, is the same as for the design-basis event, with the addition of fuel elements and subassemblies in storage. Although the electrorefiners are seismically qualified, one of the two electrorefiners in the Fuel Conditioning Facility argon cell is conservatively assumed to spill its molten salt. It is assumed that 700 kilograms (1,540 pounds) of salt is fully loaded with radionuclides and about to be replaced at the time of the accident. The damage ratio for all but the fuel assemblies in storage is assumed to be 1.0 as in the design-basis seismic event. Fuel assemblies are stored in racks with cladding intact. In the seismic event, some can be expected to fall out of the rack or be hit by falling debris, but it is not reasonable to assume all assemblies would be damaged. It is assumed that 10 percent, or 12, of the assemblies stored in the cells at the time of the seismic event experience cladding failure and release of gaseous and volatile fission products. The airborne release fraction/respirable fraction is the same as for the design-basis seismic event, with the addition of krypton and cesium from the failed EBR-II driver assemblies. The airborne release fraction/respirable fraction for krypton and tritium, both elements in the gaseous state, is 1.0. For the molten salt spill, the airborne release fraction/respirable fraction for viscous solutions (DOE 1994b) are used: 4×10^{-6} for the airborne release fraction and 0.8 for the respirable fraction. The forces associated with the beyond-design-basis earthquake are assumed to result in the failure of confinement integrity. The cells are assumed to experience major failure, and the release is directly to the atmosphere. The leak path factor is 1.0.

Table F–8 Material at Risk and Release Fraction Values Assumed for the Beyond-Basis Seismic Event at ANL-W

Accident	Material at Risk[a]			DR	ARF	RF	LPF	Source Term (curies)	
	Isotope	Blanket (Ci)	Driver (Ci)					Blanket	Driver
Beyond-design-basis seismic event and salt powder spill in the Hot Fuel Examination Facility	Sr-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	1	2.96E-03	2.10E-01
	Y-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	1	2.96E-03	2.10E-01
	I-129	8.79E-04	1.31E-02	0.01	2.0E-03	3.0E-01	1	5.27E-09	7.86E-08
	Cs-134	8.18E+00	3.13E+02	0.01	2.0E-03	3.0E-01	1	4.91E-05	1.88E-03
	Cs-137	1.06E+03	3.92E+04	0.01	2.0E-03	3.0E-01	1	6.36E-03	2.35E-01
	Ba-137M	1.00E+03	3.71E+04	0.01	2.0E-03	3.0E-01	1	6.00E-03	2.23E-01
	Ce-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	1	2.30E-04	3.16E-03
	Pr-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	1	2.30E-04	3.16E-03
	Pm-147	2.48E+02	1.47E+04	0.01	2.0E-03	3.0E-01	1	1.49E-03	8.82E-02
	Sm-151	6.10E+01	9.48E+02	0.01	2.0E-03	3.0E-01	1	3.66E-04	5.69E-03
	Eu-154	4.48E+00	1.01E+02	0.01	2.0E-03	3.0E-01	1	2.69E-05	6.06E-04
	Eu-155	2.94E+01	6.77E+02	0.01	2.0E-03	3.0E-01	1	1.76E-04	4.06E-03
	Th-228	9.46E-05	9.13E-03	0.01	2.0E-03	3.0E-01	1	5.68E-10	5.48E-08
	Np-237	7.94E-05	5.13E-02	0.01	2.0E-03	3.0E-01	1	4.76E-10	3.08E-07
	Pu-238	5.15E+00	6.68E+01	0.01	2.0E-03	3.0E-01	1	3.09E-05	4.01E-04
	Pu-239	4.13E+02	1.08E+02	0.01	2.0E-03	3.0E-01	1	2.48E-03	6.48E-04
	Pu-240	2.84E+01	3.67E+00	0.01	2.0E-03	3.0E-01	1	1.70E-04	2.20E-05
	Pu-241	1.15E+02	8.93E+00	0.01	2.0E-03	3.0E-01	1	6.90E-04	5.36E-05
	Am-241	9.95E+00	6.94E-02	0.01	2.0E-03	3.0E-01	1	5.97E-05	4.16E-07
	Am-242M	1.03E-01	5.88E-05	0.01	2.0E-03	3.0E-01	1	6.18E-07	3.53E-10
Beyond-design-basis seismic event and metal fire in the Fuel Conditioning Facility argon cell	H-3	1.42E-01	2.46E+01	1	1	1	1	1.42E-01	2.46E+01
	C-14	1.19E-03	3.98E+03	1	2.5E-06	1	1	2.99E-09	9.95E-03
	Fe-55	1.80E+00	9.74E+01	1	2.5E-06	1	1	4.51E-06	2.44E-04
	Ni-63	6.12E-02	4.58E+02	1	2.5E-06	1	1	1.53E-07	1.15E-03
	Kr-85	1.04E+00	3.78E+02	1	1	1	1	1.04E+00	3.78E+02
	Sr-90	1.61E+01	3.94E+03	1	2.5E-06	1	1	4.04E-05	9.85E-03
	Y-90	1.61E+01	3.94E+03	1	2.5E-06	1	1	4.04E-05	9.85E-03
	Ru-106	2.70E+00	3.02E+01	1	2.5E-04	1	1	6.75E-04	7.55E-03
	Rh-106	2.70E+00	3.02E+01	1	2.5E-06	1	1	6.75E-06	7.55E-05
	Cd-113M	1.42E-02	9.28E-01	1	2.5E-06	1	1	3.56E-08	2.32E-06
	Sb-125	4.62E-01	5.92E+01	1	2.5E-06	1	1	1.15E-06	1.48E-04
	I-129	1.90E-01	2.46E+01	1	2.5E-06	1	1	4.76E-07	6.15E-05
	Te-125M	2.88E-05	1.47E-03	1	1	1	1	2.88E-05	1.47E-03
	Cs-134	2.68E-01	3.52E+01	1	2.5E-04	1	1	6.70E-05	8.80E-03
	Cs-137	3.46E+01	4.42E+03	1	2.5E-04	1	1	8.65E-03	1.10E+00
	Ba-137M	3.28E+01	4.18E+03	1	2.5E-06	1	1	8.20E-05	1.05E-02
	Ce-144	1.25E+00	5.92E+01	1	2.5E-06	1	1	3.14E-06	1.48E-04
	Pr-144	1.25E+00	5.92E+01	1	2.5E-06	1	1	3.14E-06	1.48E-04
	Pm-147	8.14E+00	1.65E+03	1	2.5E-06	1	1	2.04E-05	4.13E-03
	Sm-151	2.00E+00	1.07E+02	1	2.5E-06	1	1	5.00E-06	2.67E-04
	Eu-154	1.47E-01	1.13E+01	1	2.5E-06	1	1	3.67E-07	2.84E-05
	Eu-155	9.62E-01	7.62E+01	1	2.5E-06	1	1	2.41E-06	1.91E-04
	Th-228	3.10E-06	1.03E-03	1	2.5E-06	1	1	7.75E-12	2.57E-09
	Np-237	2.66E-05	8.08E-01	1	2.5E-06	1	1	6.65E-11	2.02E-06
	U-234	7.54E-05	2.62E-02	1	2.5E-06	1	1	1.89E-10	6.55E-08
	U-235	8.48E-05	2.42E-02	1	2.5E-06	1	1	2.12E-10	6.05E-08

Accident	Material at Risk[a]			DR	ARF	RF	LPF	Source Term (curies)	
	Isotope	Blanket (Ci)	Driver (Ci)					Blanket	Driver
Beyond-design-basis seismic event and metal fire in the Fuel Conditioning Facility argon cell (cont'd)	U-236	6.54E-03	2.22E-03	1	2.5E-06	1	1	1.64E-08	5.55E-09
	U-238	2.60E-06	5.78E-03	1	2.5E-06	1	1	6.50E-12	1.45E-08
	Pu-238	1.88E-01	3.32E+00	1	2.5E-06	1	1	4.70E-07	8.30E-06
	Pu-239	1.51E+01	5.38E+00	1	2.5E-06	1	1	3.77E-05	1.35E-05
	Pu-240	1.04E+00	1.82E-01	1	2.5E-06	1	1	2.59E-06	4.56E-07
	Pu-241	4.20E+00	4.44E-01	1	2.5E-06	1	1	1.05E-05	1.11E-06
	Am-241	3.26E-01	7.82E-03	1	2.4E-06	1	1	8.15E-07	1.96E-08
	Am-242M	3.38E-03	6.62E-06	1	2.5E-06	1	1	8.45E-09	1.66E-11
Beyond-design-basis seismic event and the liquid salt spill at Fuel Conditioning Facility	Sr-90	3.45E+03	2.45E+05	1	4.0E-06	0.8	1	1.10E-02	7.84E-01
	Y-90	3.45E+03	2.45E+05	1	4.0E-06	0.8	1	1.10E-02	7.84E-01
	I-129	6.15E-03	9.17E-02	1	4.0E-06	0.8	1	1.97E-06	2.93E-05
	Cs-134	5.73E+01	2.19E+03	1	4.0E-06	0.8	1	1.83E-02	7.01E-01
	Cs-137	7.42E+03	2.74E+05	1	4.0E-06	0.8	1	2.37E+00	8.78E+01
	Ba-137M	7.00E+03	2.60E+05	1	4.0E-06	0.8	1	2.24E-04	8.31E-01
	Ce-144	2.68E+02	3.68E+03	1	4.0E-06	0.8	1	8.58E-04	1.18E-02
	Pr-144	2.68E+02	3.68E+03	1	4.0E-06	0.8	1	8.58E-04	1.18E-02
	Pm-147	1.74E+03	1.03E+05	1	4.0E-06	0.8	1	5.56E-03	3.29E-01
	Sm-151	4.27E+02	6.64E+03	1	4.0E-06	0.8	1	1.37E-03	2.12E-02
	Eu-154	3.14E+01	7.07E+02	1	4.0E-06	0.8	1	1.00E-04	2.26E-03
	Eu-155	2.06E+02	4.74E+03	1	4.0E-06	0.8	1	6.59E-04	1.52E-02
	Th-228	6.62E-04	6.39E-02	1	4.0E-06	0.8	1	2.12E-09	2.05E-07
	Np-237	5.56E-04	3.59E-01	1	4.0E-06	0.8	1	1.78E-09	1.15E-06
	Pu-238	3.61E+01	4.68E+02	1	4.0E-06	0.8	1	1.15E-04	1.50E-03
	Pu-239	2.89E+03	7.56E+02	1	4.0E-06	0.8	1	9.25E-03	2.42E-03
	Pu-240	1.99E+02	2.57E+01	1	4.0E-06	0.8	1	6.36E-04	8.22E-05
	Pu-241	8.05E+02	6.25E+01	1	4.0E-06	0.8	1	2.58E-03	2.00E-04
	Am-241	6.97E+01	4.86E-01	1	4.0E-06	0.8	1	2.23E-04	1.55E-06
	Am-242M	7.21E-01	4.12E-04	1	4.0E-06	0.8	1	2.31E-06	1.32E-09
Beyond-design-basis seismic event and stored fuel assembly cladding failure	H-3	.335	62	1	1	1	1	.335	62
	Kr-85	2.44	953	1	1	1	1	2.44	953

Ci = curies, DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
[a] Radionuclide Inventory from Appendix D.

❒ **Description of Accident Scenarios for the Melt and Dilute Process at the ANL-W Site—**The melt and dilute process would occur in the Hot Fuel Examination Facility hot cell at ANL-W. Two melt and dilute process options are considered for ANL-W: 1) melt and dilute cleaned (removed metallic sodium), blanket fuel, and 2) melt and dilute cleaned blanket fuel and cleaned (to extent possible) driver fuel (see Appendix C for more details). Metallic irons would be added to both process options to form a more corrosive resistance metal waste. Both options would occur at a temperature range of 1,000 to 1,400 °C (1,832 to 2,552 °F). For analysis purposes, it was assumed that on the average, 120 batches of melt and dilute process could be performed per year, considering an 80 percent availability and a 3-batches-per-week operation. Each batch processes about 60 kilograms (132 pounds) of heavy metal of blanket fuel or about 16 kilograms of driver fuel (diluted with depleted uranium to a 60-kilograms equivalent heavy metal). This leads to 8 years of operations for processing blanket fuels and 2 years of processing for driver fuel. Several

processes are available for cleaning of the spent nuclear fuel, the process that appears to be most practical involves the evaporation and vaporization of the sodium metal. The fuel elements would be heated to a temperature above the 200 °C (392 °F) melting point of sodium and the molten sodium drained into a collection tank. The temperature of this bulk sodium would be raised to 500 °C (932 °F), volatilizing the sodium and separating the entrained cesium from the sodium. (See Appendix C for a more detailed description of this process.)

Table F–9 identifies a list of accident scenarios that were considered to be applicable to the melt and dilute process as proposed at ANL-W. These scenarios are based on the analysis of the melt and dilute process provided in the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998). The accident scenarios and the corresponding source terms have been modified to reflect the specifics associated with the design of the facility (Hot Fuel Examination Facility), the characteristics of the fuel type being processed, the material at risk, and the related release fractions.

Table F–9 Selected Accident Scenarios Melt and Dilute at ANL-W

Scenario	*Frequency (per year)*
Nuclear Criticality	0.0003
Cask Drop	0.01
Waste Handling Accident	0.024
Sodium Fire[a]	0.008
Aircraft Crash	6×10^{-7} to 1×10^{-8}
Design Basis Earthquake	0.008

[a] This event is evaluated as being a direct consequence of the design-basis earthquake.

Each accident scenario description sets the condition of the accident and provides a summary of material involved. The following paragraphs provide a summary of the accidents analyzed, the material at risk and the release factors, for the EBR-II blanket, (the Fermi-1 blanket fuel has a very low radioactive inventory), and EBR-II driver fuel.

- *Nuclear Criticality*—A criticality accident could result from the processing of multiple batches, double batching, of fissile material to the melter. This accident was considered for the driver fuel only. The criticality was assumed to consist of 5×10^{17} fissions, based on a process criticality fission yield (DOE 1998). The Hot Fuel Examination Facility structure would not be compromised and its ventilation system would be expected to continue to function after a criticality event. Procedural controls will be used to prevent such an accident. Therefore, such an accident would be the result of a combination of human errors, as all criticality controls are designed to meet double contingency requirements. The Hot Fuel Examination Facility (ANL 1998b) identifies a criticality event as an incredible event, i.e., assigns it a frequency of less than 1×10^{-6} per year. However, this Safety Analysis Report does not specifically address melt and dilute operations. A criticality event for the melt and dilute process has been addressed for the SRS melt and dilute process (DOE 1998) and for consistency between alternatives, this analysis has been adapted. Based on the assumption of approximately 100 batching operations per year and the frequency analysis for this type of processing criticality at SRS (DOE 1998), the expected frequency of this event is 0.0003 per year to melt and dilute operations at ANL-W. The material at risk and release fractions are provided in **Table F–10**. The damage ratio and leak path factor for the volatile, gaseous fission products have conservatively been assumed to be 1.0. A respirable fraction value of 1.0 was also used. The airborne release fraction values range from 0.5 to 0.05 (DOE 1994b).

Table F–10 Melt and Dilute Process Material at Risk and Release Fraction Values for the Nuclear Criticality Event at ANL-W

Material at Risk						
Isotope	***curies***	***DR***	***ARF***	***RF***	***LPF***	***Source Term (curies)***
Br-83	4.90	1	0.05	1	1	0.25
Br-84	16.3	1	0.05	1	1	0.82
Kr-83M	1.50	1	0.5	1	1	0.75
Kr-85M	7.2	1	0.5	1	1	3.6
Kr-87	32.8	1	0.5	1	1	17
Kr-88	32.9	1	0.5	1	1	17
Kr-89	1820	1	0.5	1	1	910
Te-129	2.70	1	0.07	1	1	0.19
Te-131	57.5	1	0.07	1	1	4.0
Te-131M	0.320	1	0.07	1	1	0.022
Te-132	1.60	1	0.07	1	1	0.11
Te-133	25.7	1	0.07	1	1	1.8
Te-133M	30.3	1	0.07	1	1	2.1
Te-134	90.5	1	0.07	1	1	6.3
I-131	0.212	1	0.05	1	1	0.011
I-132	0.855	1	0.05	1	1	0.043
I-133	6.80	1	0.05	1	1	0.34
I-134	98.0	1	0.05	1	1	4.9
I-135	22.1	1	0.05	1	1	1.1
Xe-133	.026	1	0.5	1	1	0.013
Xe-135	2.61	1	0.5	1	1	1.3
Xe-135M	23.9	1	0.5	1	1	0.12
Xe-137	1940	1	0.5	1	1	0.097
Xe-138	665	1	0.5	1	1	0.033

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

- *Cask Drop*—Similar to that analyzed for the electrometallurgical treatment process, spent nuclear fuel casks will be handled frequently when the sodium-bonded fuel is processed using the melt and dilute process. (Spent nuclear fuel handling at the ANL-W site is not limited to that associated with the treatment of the sodium-bonded spent nuclear fuel. The accident discussed here is intended to address only that portion of the handling activity that can be directly attributed to the treatment of sodium-bonded spent nuclear fuel.) The accident involves a dropped cask during loading or unloading, seal failure, and spent nuclear fuel cladding failure sufficient to release gaseous and volatile fission products to the atmosphere, and is the same as previously described for the cask drop accident for the electrometallurgical treatment process. Material at risk and release fraction values are provided in Table F–4. (See the accident description for more detail.)

- *Waste Handling Accident*—The filters used in the melt and dilute off-gas exhaust system must be periodically cleaned and the resultant liquid waste disposed. The decontamination of the filters is assumed to be performed after a number of batches is processed, i.e., here it was assumed that after processing of 600 kilograms (1,320 pounds) of heavy metal blanket fuel, or 160 kilograms (352 pounds) of heavy metal driver fuel, the filters will be decontaminated. It was postulated that a spill would occur during the transfer of the decontaminated liquid from one container to another. The event frequency is estimated at 0.024 events per year (DOE 1998). The material at risk is the fission products released during the melting process and collected on the filters. This includes the fission products with boiling points at or below 1400 °C (2,552 °F) and some metal oxides that can be expected to be formed during

the heating process. (WSRC 1998b). A damage ratio of 0.5 was assumed to account for the spilling of half of the material during the accident. Airborne release fraction and respirable fraction values of 0.0002 and 0.5 respectively were chosen for the material based on the release of material from aqueous spills. (DOE 1994b) The spill is assumed to occur in an area not provided with a filtration system, and therefore, the leak path factor is 1.0. The material at risk, release fractions, and curies released for this accident for both EBR-II blanket and driver fuel are presented in **Table F–11**.

Table F–11 Melt and Dilute Process Material at Risk and Release Fraction Values for the Waste Handling Accident at ANL-W

Material At Risk							*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*	*DR*[a]	*ARF*	*RF*	*LPF*	*Blanket*	*Driver*
Sr-90	484.2	31520	0.5	0.0002	0.5	1	0.024	1.58
Sb-125	13.86	473.6	0.5	0.0002	0.5	1	0.00069	0.024
Te-125M	5.71	196.8	0.5	0.0002	0.5	1	0.00029	0.0098
I-129	0.00086	0.012	0.5	0.0002	0.5	1	4.320E-08	6.0×10^{-7}
Cs-134	8.04	281.6	0.5	0.0002	0.5	1	0.000402	0.014
Cs-137	1038	35360	0.5	0.0002	0.5	1	0.0519	1.77
Pu-238	5.63	26.6	0.000015	0.0002	0.5	1	8.4×10^{-9}	4.0×10^{-8}
Pu-239	451.8	43.0	0.000015	0.0002	0.5	1	6.8×10^{-7}	6.5×10^{-8}
Pu-240	31.08	1.5	0.000015	0.0002	0.5	1	4.7×10^{-8}	2.3×10^{-9}
Pu-241	126.0	3.6	0.000015	0.0002	0.5	1	1.9×10^{-7}	5.4×10^{-9}
Am-241	9.78	0.063	0.000015	0.0002	0.5	1	1.5×10^{-8}	9.5×10^{-11}
Am-242M	0.10	0.000016	0.000015	0.0002	0.5	1	1.5×10^{-10}	2.4×10^{-14}

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

[*] DR (damage ratio) for particulate that would not be condensed in the off-gas system includes a factor of 0.00003 to account for fraction oxidized and released from liquid metals and captured on the filters.

- *Aircraft crash*—The potential for an aircraft crash was evaluated for the Hot Fuel Examination Facility or Fuel Conditioning Facility as part of the evaluation of the electrometallurgical treatment process. The discussion provided previously is applicable to the use of the Hot Fuel Examination Facility in the melt and dilute process, see the discussion for the electrometallurgical treatment process earlier in this section. It was concluded that the likelihood of an aircraft crash causing damage to the facility process is not reasonably foreseeable, therefore, no specific analysis would be needed.

- *Fire*—The fire event selected for analysis is postulated to occur during the fuel cleaning process for the sodium-bonded spent nuclear fuel. The fire is the result of a breach in the Hot Fuel Examination Facility cell followed by a sodium fire. This event can occur as a result of the design-basis seismic event, which results in main cell breaches at piping and ventilation penetrations and results in a failure of the ventilation system. The frequency of this event is 0.008 per year.

It has been estimated that approximately 10 percent of the cesium in the spent nuclear fuel has migrated from the fuel region and bonded with the sodium being removed in the fuel cleaning process. Using the radionuclide inventories provided in Appendix D for the EBR-II driver and radial blanket elements to represent all driver and blanket fuel, this results in 670 curies of cesium-134 and 76,000 curies of cesium-137 entrained within the sodium. Assuming that as much as one half of the sodium is accumulated within the collection tank prior to processing to remove cesium from the sodium, the material at risk for the sodium fire would be 340 curies of cesium-134 and 38,000 curies of cesium-137. The release fractions selected for this accident are a damage ratio of 1.0, a combined airborne release fraction/respirable fraction of 2.5×10^{-4} and a leak path factor of 0.125. The airborne release fraction/respirable fraction is the same value as that used for cesium release from a metal fire in the design-basis seismic event analysis. The leak path factor is the value used for the Hot Fuel

Examination Facility during a design-basis seismic event. The quantity of cesium released (source term) as a result of this accident is 0.011 curies of cesium-134 and 1.2 curies of cesium-137.

- *Design Basis Seismic Event*—This is the same accident that was developed for the Design Basis Seismic Event for the electrometallurgical treatment of fuel in the Hot Fuel Examination Facility. The equipment availability and damage are assumed to be the same when the facility is used in the melt and dilute process as when it is used for the electrometallurgical treatment process. Consistent with the facility safety analysis report, the ventilation system was assumed to have failed, creating a leak path factor of 0.125. The frequency of this event is 0.008 per year (or once in 125 years).

The damage ratio, airborne release fraction, respirable fraction, and leak path factor are the same as for the electrometallurgical treatment process design-basis seismic event, with a few exceptions. Because the melt and dilute process at ANL-W operates at an elevated temperature of about 1,400 °C (2,552 °F), some fission products would boil off during the process and enter the off-gas control system. The airborne release fraction and respirable fraction for these volatilized fission product materials, e.g., strontium, antimony, cesium, tellurium, and iodine are set to 1.0 (DOE 1994b). In addition, even though some of these materials could have been condensed or not be in a vapor form (i.e., removed from the system) at the time of the accident, for conservatism it was assumed that all would be volatilized upon the initiation of the accident. The gaseous krypton and tritium are not considered here, since these are assumed to have been released to the environment during the fuel cleaning process. The source terms and release fractions are provided in **Table F–12**.

Table F–12 Material at Risk and Release Fraction Values for the Melt & Dilute Design Basis Seismic Event Accident at ANL-W

Material At Risk							*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*	*DR*	*ARF*	*RF*	*LPF*	*Blanket*	*Driver*
Sr-90	48.4	3152	1	1	1	0.125	6.05	394
Y-90	48.4	3152	1	4.0×10^{-6}	0.8	0.125	0.000019	0.0013
Ru-106	8.1	24.16	1	4.0×10^{-6}	0.8	0.125	3.2×10^{-6}	9.8×10^{-6}
Rh-106	8.1	24.16	1	4.0×10^{-6}	0.8	0.125	3.2×10^{-6}	9.8×10^{-6}
Cd-113M	0.043	0.74	1	4.0×10^{-6}	0.8	0.125	1.7×10^{-8}	3.0×10^{-7}
Sb-125	1.39	47.36	1	1	1	0.125	0.17	5.92
Te-125M	0.57	19.68	1	1	1	0.125	0.071	2.46
I-129	0.000086	0.0012	1	1	1	0.125	0.000011	0.00015
Cs-134	0.80	28.16	1	1	1	0.125	0.10	3.52
Cs-137	103.8	3536.0	1	1	1	0.125	12.98	442
Ba-137M	98.4	3344.0	1	4.0×10^{-6}	0.8	0.125	0.000039	0.0013
Ce-144	3.76	47.36	1	4.0×10^{-6}	0.8	0.125	1.5×10^{-6}	0.000019
Pr-144	3.76	47.36	1	4.0×10^{-6}	0.8	0.125	1.5×10^{-6}	0.000019
Pm-147	24.4	1321.6	1	4.0×10^{-6}	0.8	0.125	9.8×10^{-6}	0.00053
Sm-151	6.0	85.44	1	4.0×10^{-6}	0.8	0.125	2.4×10^{-6}	0.000034
Eu-154	0.44	9.07	1	4.0×10^{-6}	0.8	0.125	1.8×10^{-7}	3.6×10^{-6}
Eu-155	2.89	60.96	1	4.0×10^{-6}	0.8	0.125	1.2×10^{-6}	0.000024
Pu-238	0.56	2.66	1	4.0×10^{-6}	0.8	0.125	2.2×10^{-7}	1.1×10^{-6}
Pu-239	45.18	4.30	1	4.0×10^{-6}	0.8	0.125	0.000018	1.7×10^{-6}
Pu-240	3.11	0.15	1	4.0×10^{-6}	0.8	0.125	1.2×10^{-6}	6.0×10^{-8}
Pu-241	12.6	0.36	1	4.0×10^{-6}	0.8	0.125	5.0×10^{-6}	1.4×10^{-7}
Am-241	0.98	0.0063	1	4.0×10^{-6}	0.8	0.125	3.9×10^{-7}	2.5×10^{-9}
Am-242M	0.010	1.6×10^{-6}	1	4.0×10^{-6}	0.8	0.125	4.0×10^{-9}	6.4×10^{-13}

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

F.2.2.1.3 Accident Scenarios Description and Source Terms at the Savannah River Site

- **Description of Accident Scenarios for the PUREX Process at SRS—**The following facilities would be used to store or process sodium-bonded fuel at SRS: F-Canyon, FB-Line, and the plutonium storage facility. The F-Canyon, FB-Line, and plutonium storage facility are part of the Building 221-F (or F-Canyon) structure. Shipments of the declad and cleaned sodium-bonded spent nuclear fuel cannot be received directly at the F-Canyon facility. The facility is not equipped to handle the transportation casks being used. The shipments would be received at the L-Reactor dissassembly basis, transferred to casks suitable for shipment to F-Canyon, and then moved to F-Canyon. The PUREX process can be used to separate the plutonium from the sodium-bonded blanket assemblies. In the PUREX process, the declad and cleaned blanket fuel would be dissolved in the F-Canyon dissolvers, and fission products would be separated from uranium and plutonium. The plutonium solution then would be pumped to the FB-Line for purification and solidification. The depleted uranium solution would be pumped to A-Line tanks for storage and future processing into depleted uranium oxides.

The accident scenarios, identified in **Table F–13** and defined in the following paragraphs, are applicable to the processing facilities as a whole (i.e., F-Canyon and FB-Line). Transfer and storage accidents were also considered for the analysis of F-Canyon related activities. The sodium-bonded spent nuclear fuel is declad and cleaned prior to shipment from ANL-W. This process results in the release of gases in the gap between the fuel and cladding (see Appendix E), the dominant radionuclides considered during the analysis of transfer (fuel and cask drop) accidents. Therefore, the accidents were not quantified. Accidents associated with the storage of the sodium-bonded spent nuclear fuel and the storage of the process products (plutonium and various waste forms) were assessed as having no additional impacts beyond those associated with the process-related accidents.

Table F–13 Selected Accident Scenarios for PUREX Process at SRS

Scenario	*Frequency (per year)*
Explosion: Ion exchange column	1×10^{-4}
Nuclear Criticality [a]	1×10^{-4}
Fire	6.1×10^{-5}
Earthquake (DBE)	1.3×10^{-4}
Aircraft Crash	less than 10^{-7}

DBE = design (evaluation) basis earthquake

[a] Only plutonium criticalities are evaluated. The potential for an americium criticality was considered but dismissed because of the limited americium mass and purity.

- *Explosion*—An explosion in an ion exchange column in the FB-Line is postulated to result from a strong exothermic reaction between nitric acid and the base resin in the cation (or anion) exchange column during plutonium solution exchange. This would result in a thermally induced pressure failure of the ion exchange vessel, and the resulting shrapnel would damage the product run tank and the product hold tank for this ion exchange pair. The explosion would breach the hot cell confinement. The plutonium in nitrite solution in the run and hold tanks would spill onto the cabinet floor and boil due to a subsequent resin fire. Based on the assumptions that the column was at its maximum load before the explosion, and the maximum quantity of liquid at the maximum allowable concentration was present, the estimated release of plutonium through the sand filter and the stack was calculated to be 0.241 grams of plutonium (DOE 1993b). No other source term is applicable to the FB-line accident. Processing in the F-Canyon removes all other fission products before the plutonium is processed in the FB-line. The frequency of such an event is estimated to be 1×10^{-4} per year (DOE 1993b).

- *Fire*—In the F-Canyon safety analysis report (WSRC 1994), a maximum fire was postulated to occur in the plutonium recycle process. The frequency of such a fire was estimated at 6.1×10^{-5} per year (WSRC 1994). The accident was assumed to burn the contents of the largest tank. The material at risk is 86,700 kilograms

(191,000 pounds) of solution. The combined [airborne release fraction][respirable fraction] was estimated to be 1×10^{-2} (DOE 1994b). The airborne materials would pass through sand filter, with a leak path factor of 0.005, before entering the atmosphere. The maximum recycle fire in the F-Canyon would result in the bounding source term (**Table F–14** gives the source term). Fire in the FB-Line would result in consequences that are several times lower than those from the F-Canyon fire.

Table F–14 Maximum Fire Source Term

Isotope	*Source Term (curies)*
Sr-90	1.5
Ru-106	12
Ce-144	36
U-234	3.0×10^{-7}
U-235	4.8×10^{-6}
U-236	4.9×10^{-6}
U-238	0.00044
Pu-238	0.19
Pu-239	1.6
Pu-240	0.36
Pu-241	4.2
Pu-242	0.000053
Am-241	0.32

- *Criticality*—A plutonium solution criticality was postulated. The criticality was assumed to consist of an initial burst of 1×10^{18} fissions in 0.5 seconds, followed at 10-minute intervals for the next 8 hours by bursts of 2×10^{17} fissions, for a total of 1×10^{19} fissions as specified in the U.S. Nuclear Regulatory Commission's Regulatory Guide 3.35 (NRC 1979) and NUREG-1320 (NRC 1988) and in the DOE-HDBK-3010-YR (DOE 1994b) report. The 10^{19} fission yield was based on the assumptions that the solution criticality occurred in a tank with a minimum volume of 3,785 liters (1,000 gallons) and that approximately 100 liters (26 gallons) of this volume evaporated due to heat released during the fission process. Based on the data provided in the DOE Safety Survey Report (DOE 1993c), a 10^{19} criticality event in the FB-Line process would result in the bounding source term (**Table F–15** gives the source terms). The frequency of such an event was estimated to be 1×10^{-4} per year.

Table F–15 Criticality Source Term for 10^{19} Fissions in Plutonium Solution

	Radioactivity (curies) [a]					
Isotope	*0-30 minutes*	*30 min-8 hours*	*Total*	*ARF* [b]	*LPF* [c]	*Source Term (curies)*
Kr-83m	15	95	110	1	1	110
Kr-85m	9.9	61	70.9	1	1	70.9
Kr -85	0.00012	0.00072	0.00084	1	1	0.00084
Kr-87	60	370	430	1	1	430
Kr-88	32	200	232	1	1	232
Kr-89	1,800	11,000	12,800	1	1	12,800
Xe-131m	0.014	0.086	0.1	1	1	0.1
Xe-133m	0.31	1.9	2.21	1	1	2.21
Xe-133	3.8	23	26.8	1	1	26.8
Xe-135m	460	2,800	3,260	1	1	3,260
Xe-135	57	350	407	1	1	407
Xe-137	6,900	42,000	48,900	1	1	48,900
Xe-138	1,500	9,500	11,000	1	1	11,000
I-131	1.5	9.5	11	0.25	1	2.75

Isotope	Radioactivity (curies) [a]			ARF [b]	LPF [c]	Source Term (curies)
	0-30 minutes	30 min-8 hours	Total			
I-132	170	1,000	1,170	0.25	1	293
I-133	22	140	162	0.25	1	40.5
I-134	600	3,700	4,300	0.25	1	1,080
I-135	63	390	453	0.25	1	113
Pu-238 [c, d]			3.6	0.0005	0.005	0
Pu-239 [c, d]			170	0.0005	0.005	0.00043
Pu-240 [c, d]			39	0.0005	0.005	0.0001
Pu-241 [c, d]			2,400	0.0005	0.005	0.006
Pu-242 [c, d]			0.003	0.0005	0.005	7.50×10^{-9}

ARF = airborne release fraction, LPF = leak path factor

[a] Regulatory Guide 3.35 (NRC 1979).

[b] Airborne release fractions are equal to 1.0 for noble gases, 0.25 for iodine, and 0.0005 for plutonium; all particles are assumed to be in the respirable range (i.e., Respirable Fraction = 1.0).

[c] Plutonium in 100 liters of solution.

[d] This plutonium is assumed to be released to the atmosphere through a high-efficiency particulate air filter (e.g., SRS's sand filter) with a 0.995 efficiency. The plutonium values are the maximum solution concentration in the FB-Line (DOE 1993b).

- *Earthquake*—Recent analyses of earthquake hazards at F-Canyon indicate that a 0.24-g peak ground acceleration level earthquake—with a return period of 8,000 years (or a frequency of 1.25×10^{-4} per year) for the F-Canyon facility—could damage the structure and cause localized interior failures as well as interior and exterior wall cracks (DOE 1996b). Previous analyses of earthquake hazards at F-Canyon estimated the consequences of such a magnitude earthquake with a higher frequency of occurrences—2×10^{-4} per year (DOE 1995b and WSRC 1994). Using the assumptions in the F-Canyon Facility Safety Analysis Report (WSRC 1994), a bounding source term was developed for an earthquake accident (**Table F–16** gives the F-Canyon source term). Given an earthquake, it was assumed that the plutonium contents in all the processes (F-Canyon and FB-Line) would be spilled on the canyon floor. It was further assumed that the airborne material would enter the environment through the building cracks, which are formed by the loss of sealant between the sections because of differential motion of the section, with a penetration leak path factor of 0.10. For the FB-Line, the material at risk is assumed to be 2,000 grams (4.4 pounds) of plutonium in a molten metal form and 2,000 grams (4.4 pounds) of plutonium in a liquid form. The airborne release fraction, times the respirable fraction is 0.0022 for the molten metal form, and 0.000047 for liquid form, including both the initial and resuspended airborne release fraction multiplied by respirable fraction values. This results in an FB-Line earthquake source term of 0.45 grams of plutonium released to the environment.

Table F–16 Maximum Earthquake Source Term

Isotope	Source Term (curies)	Isotope	Source Term (curies)
Sr-90	0.156	Pu-239	0.168
Ru-106	127	Pu-240	0.0387
Ce-144	3.72	Pu-241	0.429
Cs-137	0.00574	Pu-242	7.18×10^{-6}
Eu-154	0.0338	Am-241	0.0151
Np-237	5.84×10^{-8}	Am-242m	0.0000630
Np-239	0.0116	Am-243	0.00616
U-234	4.09×10^{-7}	Cm-244	0.668
U-235	5.07×10^{-7}	Cm-245	0.0000542
U-236	5.12×10^{-7}	Cm-246	0.0000844
U-238	0.0000457	Cm-247	4.10×10^{-10}
Pu-238	0.0276		

- *Aircraft Crash*—The location of the F-Canyon facility is far away from any airport; therefore, no takeoff and landing crash accidents need to be considered. The crashes that could occur during in-flight would need to be considered. According to the DOE Standard on aircraft crash analysis, DOE-STD-3014-96 (DOE 1996c), the expected crash frequency for the site is approximately 2×10^{-4} per square-mile per year from general aviation, 6×10^{-7} and 2×10^{-6} per square-mile per year from air carrier and air taxies, respectively, and 1×10^{-7} and 6×10^{-7} per square-mile per year from large military and small military aircraft, respectively. Using the building dimensions and the data provided in the DOE Standard for aircraft crash analysis, an upper bound frequency for an aircraft crash into the canyon buildings was estimated to be 4.6×10^{-6} and 1.5×10^{-7} per year for general aviation and commuter (air taxi) aircraft, respectively. These values were calculated without considering any site-specific effects (e.g., the topography and building structures around the facility). Considering the available skid distance of 60 meters (200 feet) that an aircraft could skid before hitting the building, the frequency of an air taxi crashing into the building would be less than 10^{-8} per year. When only crashes that directly hit the structure were considered, general aviation aircraft would have the only estimated crash frequency greater than 10^{-7} per year. The F-Canyon building is a maximum resistant construction structure designed to withstand a pressure of 47.9 kilopascal (1,000 pounds per square foot). Therefore, crashes of small aircraft (helicopter or a small observation/ security aircraft) into these buildings are not expected to damage the buildings. If a general aviation aircraft were to crash into the buildings, its consequences (both in magnitude and frequency) would be smaller than that hypothesized for a design-basis evaluation earthquake.

❒ **Description of Accident Scenarios for the Melt and Dilute at SRS—**The following accidents have been considered for the melt and dilute option, when performed at the Building 105-L (after receipt of the declad and cleaned spent nuclear fuel at the L-Reactor Disassembly Basin) as proposed in the SRS Spent Nuclear Fuel Management Environmental Impact Statement (DOE 1998). In this process, the declad and cleaned blanket fuel along with aluminum metal would be heated to approximately 1,000 °C (1,832 °F) to form an alloy of 30 percent uranium and 70 percent aluminum and cast as ingots. The heating process would remove some of the radionuclides found in the spent nuclear fuel. The analysis assumes a batch size of 60 kilograms (132 pounds) of heavy metal, which is the batch size limit for this process when performed in Building 105-L. The radionuclide content of an EBR-II radial blanket fuel has been conservatively used to represent the radionuclide content of all blanket fuels. The accident scenarios identified in **Table F–17**, and described in the following paragraphs, are applicable to the melt and dilute processing of the blanket fuel in SRS Building 105-L. Accidents associated with the onsite transfer and storage of the declad and cleaned spent nuclear fuel were considered for analysis. As in the accident analysis for the PUREX process, these accidents were not quantified. Accidents associated with the transfer and storage of the spent nuclear fuel and diluted waste forms were assessed as having no additional impacts beyond those analyzed for process-related accidents.

Table F–17 Selected Accident Scenarios Melt and Dilute at SRS Building 105-L

Scenario	*Frequency (per year)*
Loss of Cooling Water	0.05
Waste Handling Spill	0.024
Loss of Electric Power	0.006
Fire	0.075
Design Basis Earthquake	NA*

* Building 105-L and the melt and dilute components are expected to remain functioning after a design-basis earthquake. The most significant impact of this event is a potential loss of offsite power. The consequences of an earthquake up to a design-basis event are thereby bounded by the loss of power event. The loss of power event has a higher frequency than the design-basis earthquake and is used in place of the design-basis earthquake.

- *Loss of Cooling Water*—The postulated melter explosion event results from a buildup or addition of impurities to the metal melt. Impurities range from water (causing a steam explosion) to chemical contaminants (possible high temperature exothermic reactions). As a result of the reaction in the metal melt, molten material is ejected from the melter into the processing structure. Cooling water pipes within the

process area could be ruptured as a result of contact with the ejected material. Should this occur, the water released would be converted to steam and would be expected to overwhelm the exhaust fans resulting in the failure of the exhaust system and an unfiltered release. The frequency of this event has been estimated to be bound by a value of 0.05 per year. (WSRC 1998a).

The material at risk is conservatively estimated to be the full radionuclide content of one melt batch. The metal melt explosion is assumed to affect approximately one-half of the material in the melter, resulting in a damage ratio of 0.5 for all material, except for the volatile gaseous fission products at 1,000 °C (1,832 °F) which are assigned a damage ratio of 1.0. The airborne release fraction and respirable fraction values of 0.001 and 0.1 are based on the conservative assumption that the characteristics of the released material will be bounded by the characteristics of material released in an explosion involving powder material (DOE 1994b). With the failure of the off-gas system, a leak path factor of 0.1 is assumed for all materials. The material at risk and release fractions are summarized in **Table F–18**.

Table F–18 Melt and Dilute Material At Risk and Release Fractions for the Loss of Cooling Water Integrity at Building 105-L

Isotope	*MAR (curies)*	*DR*	*ARF* [a]	*RF*	*LPF*	*Source Term*
Fe-55	5.41	0.5	0.001	0.1	0.1	2.71e-05
Ni-63	0.184	0.5	0.001	0.1	0.1	9.20e-07
Sr-90	48.4	0.5	0.001	0.1	0.1	2.42e-04
Y-90	48.4	0.5	0.001	0.1	0.1	2.42e-04
Ru-106	8.1	0.5	0.001	0.1	0.1	4.05e-05
Rh-106	8.1	0.5	0.001	0.1	0.1	4.05e-05
Cd-113M	0.0427	0.5	0.001	0.1	0.1	2.14e-07
Sb-125	1.39	0.5	0.001	0.1	0.1	6.95e-06
Te-125M	0.571	1	1	1	0.1	0.0571
I-129	0.000086	1	1	1	0.1	8.64×10^{-6}
Cs-134	0.804	1	1	1	0.1	0.0804
Cs-137	104	1	1	1	0.1	10.4
Ba-137M	98.4	0.5	0.001	0.1	0.1	4.92e-04
Ce-144	3.76	0.5	0.001	0.1	0.1	1.88e-05
Pr-144	3.76	0.5	0.001	0.1	0.1	1.88e-05
Pm-147	24.4	0.5	0.001	0.1	0.1	1.22e-06
Sm-151	6	0.5	0.001	0.1	0.1	3.00e-05
Eu-154	0.44	0.5	0.001	0.1	0.1	2.20e-06
Eu-155	2.89	0.5	0.001	0.1	0.1	1.45e-05
Th-22	9.30e-06	0.5	0.001	0.1	0.1	4.65e-11
U-234	0.00008	0.5	0.001	0.1	0.1	3.99e-10
U-235	0.000226	0.5	0.001	0.1	0.1	1.13e-09
U-236	0.000254	0.5	0.001	0.1	0.1	1.27e-09
U-238	0.0196	0.5	0.001	0.1	0.1	9.80e-08
Np-237	7.80e-06	0.5	0.001	0.1	0.1	3.90e-11
Pu-238	0.563	0.5	0.001	0.1	0.1	2.82e-06
Pu-239	45.2	0.5	0.001	0.1	0.1	2.26e-04
Pu-240	3.11	0.5	0.001	0.1	0.1	1.56e-05
Pu-241	12.6	0.5	0.001	0.1	0.1	6.30e-05
Am-241	0.978	0.5	0.001	0.1	0.1	4.89e-06
Am-242M	0.0101	0.5	0.001	0.1	0.1	5.05e-08

MAR = material at risk, DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

[a] The airborne release fraction values provide here are conservatively based on the assumption that reaction would behave as explosive powder mix.

- *Waste Handling Accident*—The filters used in the melt and dilute off-gas exhaust system must be periodically cleaned and the resultant liquid waste disposed. The decontamination of the filters is assumed to be performed after a number of batches is processed, i.e., here it was assumed that after processing 600 kilograms, 10 batches, of heavy metal blanket fuel, the filters will be decontaminated. It was postulated that during the transfer of the decontaminate liquid from one container to another, a spill would occur. The event frequency is estimated at 0.024 per year (DOE 1998). The material at risk is the fission products released during the melting process and collected on the filters. This includes the fission products with boiling points at or below 1,000 °C (1,832 °F) and some metal oxides that can be expected to be formed during the heating process. (WSRC 1998b). A damage ratio of 0.5 was assumed to account for the spilling of half of the material during the accident. Airborne release fraction and respirable fraction values of 0.0002 and 0.5 respectively were chosen for the material based on the release of material from aqueous spills (DOE 1994b). The spill is assumed to occur in an area not provided with a filtration system, and therefore, the leak path factor is 1.0. These material at risk, release fractions, and curies released for this accident for both EBR-II blanket and driver fuel are presented in **Table F–19**.

Table F–19 Melt and Dilute Material At Risk and Release Fractions for the Waste Handling Accident at Building 105-L

Isotope	*MAR*	*DR**	*ARF*	*RF*	*LPF*	*Source Term*
Te-125M	5.71	0.5	0.0002	0.5	1	0.000286
I-129	0.000864	0.5	0.0002	0.5	1	4.32×10^{-8}
Cs-134	8.04	0.5	0.0002	0.5	1	0.000402
Cs-137	1040	0.5	0.0002	0.5	1	0.052
Pu-238	0.563	0.000015	0.0002	0.5	1	8.45×10^{-9}
Pu-239	45.2	0.000015	0.0002	0.5	1	6.78×10^{-7}
Pu-240	31.1	0.000015	0.0002	0.5	1	4.67×10^{-8}
Pu-241	126	0.000015	0.0002	0.5	1	1.89×10^{-7}
Am-241	9.78	0.000015	0.0002	0.5	1	1.47×10^{-8}
Am-242M	0.101	0.000015	0.0002	0.5	1	1.52×10^{-10}

MAR = material at risk, DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
* Damage ratios for neptunium, plutonium, and americinium include an airborne release fraction value of 0.00003 to account for fraction released from liquid metals and captured on the filters.

- *Loss of Offsite Power*—The loss of offsite power, with the subsequent failure of the onsite power supply, will result in the failure of the off-gas system, and a potential unfiltered release path to the environment. The probability of this combination of events is conservatively estimated at 0.006 per year (WSRC 1998a). The material at risk is assumed to be the volatile radionuclide inventory of one processing batch of material (approximately 60 kilograms [132 pounds] of heavy metal). Additionally, some amounts of radioactive metallic and metallic oxide dusts could be generated and released during a loss of power event. The airborne release fraction/respirable fraction values for the gaseous fission products are assumed to be 1.0, while the metallic dust release fractions at elevated temperatures are an airborne release fraction of 0.00003 and respirable fraction of 0.04 (DOE 1994b). A leak path factor of 0.5 has been used for all material to account for possible plate out during migration of material out of the processing area. The material at risk and release fraction data are summarized in **Table F–20**.

Table F–20 Melt and Dilute Material At Risk and Release Fractions for the Loss of Power at Building 105-L

Isotope	*MAR*	*DR*	*ARF*	*RF*	*LPF*	*Source Term*
Te-125M	0.571	1	1	1	0.5	2.86e-01
I-129	0.000086	1	1	1	0.5	4.32e-05
Cs-134	0.804	1	1	1	0.5	4.02e-01
Cs-137	104	1	1	1	0.5	5.20e+01
Pu-238	0.563	1	0.00003	0.04	0.5	3.38e-07
Pu-239	45.2	1	0.00003	0.04	0.5	2.71e-05
Pu-240	3.11	1	0.00003	0.04	0.5	1.87e-06
Pu-241	12.6	1	0.00003	0.04	0.5	7.56e-06
Am-241	0.978	1	0.00003	0.04	0.5	5.87e-07
Am-242M	0.0101	1	0.00003	0.04	0.5	6.06e-09

MAR = material at risk, DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

- *Area Fire*–Fires in Building 105-L have the potential to release material from several different sources. Fires have the potential to release material from the chemical decontaminate solution and the off-gas filters and baffles, and have the potential to affect the ventilation and filtration system resulting in the release modeled for the loss of power event. The fire selected for analysis is the fire that results in the failure of the waste container and releases some of the decontaminate solution. This fire has the potential to release more material than a fire that impacts the off-gas filters and baffles. The frequency of a fire in Building 105-L, based on site-wide fire data for SRS, is 0.075 fires per year. This frequency has been conservatively used as the frequency of a fire that impacts the chemical decontaminate solution. The material at risk is the same as for the waste handling accident, the volatile gases, and metallic and metallic oxide dust that is the result of the processing of 10 batches of material in the melter. All material in the waste container is at risk and the damage ratio is assumed to be 1.0. Boiling of a shallow pool of aqueous solution results in bounding airborne release fraction and respirable fraction values of 0.002 and 1 respectively (DOE 1994b). No credit is taken for any reduction due to leak path factor, i.e., a leak path factor of 1.0 is used. **Table F–21** summarizes the material at risk and release fractions for this accident scenario.

Table F–21 Melt and Dilute Material At Risk and Release Fractions for the Area Fire at Building 105-L

Isotope	*MAR*	*DR*	*ARF*	*RF*	*LPF*	*Source Term*
Te-125M	5.71	1	0.002	1	1	0.0114
I-129	0.0006	1	0.002	1	1	1.73×10^{-6}
Cs-134	8.04	1	0.002	1	1	0.0161
Cs-137	1040	1	0.002	1	1	2.08
Np-237	7.80×10^{-5}	0.00003	0.002	1	1	4.68×10^{-12}
Pu-238	5.63	0.00003	0.002	1	1	3.38×10^{-7}
Pu-239	452	0.00003	0.002	1	1	0.0000271
Pu-240	31.1	0.00003	0.002	1	1	1.87×10^{-6}
Pu-241	126	0.00003	0.002	1	1	7.56×10^{-6}
Am-241	9.78	0.00003	0.002	1	1	5.87×10^{-7}
Am-242M	0.101	0.00003	0.002	1	1	6.06×10^{-9}

MAR = material at risk, DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

F.2.2.2 Consequences and Risk Calculations

Once the source term for each accident scenario is determined, the radiological consequences are calculated. The calculations vary depending on how the release is dispersed, what material is involved, and which receptor is being considered. Risks are calculated based on the accident's frequency and its consequences. The risks are also stated in terms of additional cancer fatalities resulting from a release using a conversion factor of 5×10^{-4} latent cancer fatalities per person-rem for the members of the public, and 4×10^{-4} latent cancer fatalities per person-rem for workers.

Radiological consequences to four different receptors are evaluated: a maximally exposed offsite individual (an individual member of the public), general population, noninvolved worker (or a co-located worker), and facility worker. The consequences to the facility workers are qualitatively evaluated. For the other receptors, quantitative estimates of consequences are made; two types of dispersion conditions are considered—95th percentile and 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; it is defined as that condition not exceeded more than 5 percent of the time. Both dispersion conditions are modeled using the GENII program, which determines the desired condition from the site-specific meteorological data in the form of a joint frequency distribution. Joint frequency data are usually produced from at least three consecutive years of site weather data in terms of percentage of time that the wind blows in specific directions (e.g., south, south-southwest, southwest) for the given midpoint (or average) wind speed class and atmospheric stability.

Radiological consequences to a receptor from an accident in the FB-line are estimated based on a calculated 50-year committed dose factor (dose factor), resulting from releases of 1 gram of plutonium with an isotopic distribution associated with the EBR-II blanket fuel (**Table F–22**). This is done because the FB-line only processes plutonium already separated in the F-Canyon.

The values given in this table represent the maximum dose to the receptor and are obtained using the GENII program.

Table F–22 Receptors' Dose Factors for Accidental Releases of 1 gram Plutonium from Accident Initiated in FB-Line

Receptor		***95 Percent Meteorological Condition***	***50 Percent Meteorological Condition***
Maximally Exposed Individual (rem)	elevated release	0.027	Not Applicable
	ground release	0.13	Not Applicable
Population (person-rem)	elevated release	1500	220
	ground release	5000	270
Worker (rem)	elevated release	Not Applicable	0.080
	ground release	Not Applicable	2.0

The consequences to involved workers are qualitatively assessed. This approach is used for two reasons: first, no adequate method exists for calculating meaningful consequences at or near the location where the accident occurs. Second, safety assurance for facility workers is demonstrated by both the workers' training and by the establishment of an Occupational Safety and Health Administration process safety management system (29 CFR 1910.119), the evaluations required by such a system, and the products derived from such evaluations (e.g., procedures, programs, emergency plans).

The consequences to the involved worker, presented in **Tables F-23** and **F–24**, are accident dependent and site-specific. In facilities where the involved worker activities include remote operations, the consequences of accidents would be lower than in facilities where the workers are near the process. The following paragraphs summarize the various potential consequences to the involved workers from the hypothesized accidents at

different sites. Additionally, a limited number of fatalities could occur in an indirect or secondary manner—for example, the involved worker could be killed by an earthquake or explosion.

Table F–23 Involved Worker Consequences from Various Hypothesized Accidents

Accident	*Consequences*
Explosion (Ion Exchange)	Could potentially result in fatal injuries (nonradiological) to the nearby involved workers. (SRS only)
Criticality	Could potentially result in fatal dose to the nearby involved workers. (Worker location outside cells [e.g., outside argon cell at ANL-W] provides worker protection)
Fire	No fatality is expected, some nearby workers could inhale the dispersed radioactive materials before using respirator and leaving the area.
Earthquake	No fatality is expected.
Spill	Nearby workers could inhale the dispersed radioactive materials before using respirator and leaving the area.

Table F–24 Involved Worker Summary

Accident Description		
SRS—PUREX Process	*F-Canyon and F-B Line*	*ANL-W*
Earthquake	47	50
Explosion, Ion Exchange Column	16	Not Applicable
Nuclear Criticality	16	15
Fire	16	4

- **Explosion**—An explosion could result in serious, even fatal, injuries to involved workers from the accident itself (at SRS). Some of the involved workers could inhale the dispersed radioactive material before using their respirators and evacuating the area. No fatality is expected from the radiological consequences.

- **Fire**—Involved workers could inhale some radioactive material, before evacuating the area. No fatality is expected from the radiological consequences.

- **Spill**—Depending on the location of the spill, nearby workers may inhale the airborne radioactive materials before evacuating the area. Involved workers normally would be wearing respirators when handling the radioactive material containers. No fatality is expected to result from such an accident.

- **Earthquake**—Involved workers could receive lethal injuries from the accident itself. No fatality is expected from radiological consequences.

- **Aircraft Crash**—Consequences similar to those of an earthquake may result from the accident.

- **Criticality**—Involved workers could receive substantial, or potentially fatal, doses from prompt neutrons and gamma rays emitted from the first pulse. After the initial pulse, the workers would evacuate the area immediately on the initiation of the criticality monitoring alarms.

Analysis Conservatism and Uncertainty

To assist in evaluating the impact of the processing options at SRS and ANL-W on a common basis, a spectrum of generic accidents were postulated for each process location. The accident scenarios were based on similar accidents documented in various site documents. When required, accident assumptions were modified to enable comparison between the sites. In cases where similar accidents were evaluated in site specific documents, the more conservative analysis assumptions were used for all sites to normalize the results for the purpose of comparison. The following accident analysis parameters have a major impact on accident consequence

estimates (i.e., dose to the public and worker): weather conditions existing at the time of the accident, material at risk, isotopic breakdown of the material at risk, and source term released to the environment.

Weather conditions assumed at the time of the accident have a large impact on dose estimates. Accident impacts to the public (both the maximally exposed individual and the population) presented in this appendix were estimated using both 95th percentile and median 50th percentile weather conditions. The impacts presented in the body of the EIS are based on the 50th percentile weather conditions for population dose (NRC 1976), and 95th percentile weather conditions (NRC 1982) for the maximally exposed individual dose (which provides conservative maximally exposed individual dose estimates). The GENII computer code was used to calculate doses to the public within 80 kilometers (50 miles) of the accident release point. The code calculates the public dose in each of 16 sectors centered at the accident release point. The GENII computer code also assumes that total source term is released into each sector and that there is no change in the weather (i.e., wind direction, wind speed, and stability class) while the accident plume is traversing the 80-kilometer sector. The use of the 95th percentile weather data rather than the expected or median 50th percentile weather data, was considered to be unrealistic for estimating population dose. Meteorological conditions used in the analysis are based on measured weather data at the site. The 95th percentile represents a very stable site meteorological condition, which cannot be expected to be applicable for a wide area up to 80 kilometers from the site. Therefore, the 50th percentile, which represent a more neutral weather condition, is more representative of expected weather conditions over a wide area.

Uncertainties in accident frequencies do not impact the accident consequences, but do impact accident risk. The site/facility specific accident frequencies (i.e., earthquake induced building damage and aircraft crash) were based on data provided by the sites. Process specific accident frequencies were estimated based on analyses provided in site specific documentation. In cases where similar accidents were evaluated in site specific documents, the more conservative accident frequency was used for all sites to normalize the results for the purpose of comparison.

Due to the layers of conservatism built into the accident analysis for the spectrum of postulated accidents, the estimated consequences and risks to the public represent the upper limit for the individual classes of accidents. The uncertainties associated with the accident frequency estimates are enveloped by the analysis conservatism.

F.2.3 Accident Analyses Consequences and Risk Results

F.2.3.1 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 interior of the fuel elements) except for stabilization activities that may be necessary for continued safe and secure storage indefinitely or until a new treatment technology is developed. Under the electrometallurgical demonstration project, approximately 0.6 metric tons of heavy metal of EBR-II driver fuel and 1 metric ton of heavy metal blanket fuel would be processed. This EIS evaluates the impacts associated with activities required to clean up and stabilize any residual waste materials generated during the Electrometallurgical Treatment Demonstration Project at ANL-W. In addition, at the completion of the project, any remaining sodium-bonded spent nuclear fuel in the process facilities would be packaged and transferred to dry storage in the Radioactive Scrap and Waste Facility. Spent nuclear fuel transfer activities and waste processing activities would be completed in about two years after equipment installation. Some of the spent nuclear fuel handling and processing accidents identified under Alternative 1 are applicable to the No Action Alternative. **Tables F-25** and **F-26** provide the dose calculation results for the design-basis and beyond-design-basis seismic events for stabilizing the residual waste. The data for the remaining accidents considered for the No Action Alternative (the salt powder spill in the Hot Fuel Examination Facility, the cask drop, and transuranic waste fire) are provided in the discussion of Alternative 1 "electrometallurgical treatment at ANL-W." Data is provided for consequences and risks to the maximally exposed offsite individual, a noninvolved worker, and the general population. The accident assumptions and parameters used in developing this information have been provided in Section 2.2 of this appendix. EBR-II driver fuel characteristics (radionuclide compositions), which bound the consequences, were used to represent the consequences and risks during stabilization of wastes

for the demonstration project for the No Action Alternative. The transuranic waste fire accident was analyzed using a generic transuranic waste package composition, rather than the driver-specific composition.

Table F–25 Summary of Dose Calculation Results for Design-Basis Seismic Events (Driver)

			95th-percentile meteorology			*50th-percentile meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (millirem)***	***Worker (millirem)***	***Population (person-rem)***	***Average individual (mrem)***
Design-Basis Events	0.008	Dose per event	11.9	52.4	0.631	0.638	4.66	1.38	00166
		Dose per year	0.0012	0.0052	6.31×10^{-5}	6.38×10^{-5}	4.66×10^{-4}	1.4×10^{-4}	1.66×10^{-6}
		LCF	6.0×10^{-10}	2.6×10^{-5}	3.2×10^{-11}	3.2×10^{-11}	1.9×10^{-10}	6.9×10^{-8}	8.3×10^{-13}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–26 Summary of Dose Calculation Results for Beyond-Design-Basis Seismic Events (Driver)

			95th-percentile meteorology			*50th-percentile meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (millirem)***	***Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Beyond-Design-Basis Events[a]	0.00001	Dose per event	95.6	41.9	5.05	5.11	37.3	11	0.133
		Dose per year	9.6×10^{-4}	4.2×10^{-4}	5.1×10^{-5}	5.1×10^{-5}	3.7×10^{-4}	1.1×10^{-4}	1.3×10^{-6}
		LCF	4.8×10^{-10}	2.1×10^{-7}	2.6×10^{-11}	2.6×10^{-11}	1.5×10^{-10}	5.5×10^{-8}	6.5×10^{-13}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

[a] During stabilization of the demonstration project waste, only the Hot Fuel Examination Facility salt powder spill would be applicable.

F.2.3.2 Alternative 1 - Electrometallurgical Treatment at ANL-W

The processing technology considered for this alternative consists solely of the electrometallurgical treatment processing of the sodium-bonded spent nuclear fuel at ANL-W, using the Fuel Conditioning Facility and Hot Fuel Examination Facility. **Tables F–27** through **F–37** provide the dose calculation results for the electrometallurgical treatment related accidents at the ANL-W facility. Data is provided for consequences and risks to the maximally exposed individual, an uninvolved worker and the general population. The accident assumptions and parameters used in developing this information has been provided in Section 2.2 of this appendix. EBR-II driver fuel and EBR-II blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively. The transuranic waste fire accident was analyzed using a generic transuranic waste package composition, rather than either a blanket or driver specific composition.

Table F–27 Summary of Dose Calculation Results for Salt Powder Spill (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Hot Fuel Examination Facility Salt Powder Spill	0.01	Dose per event	4.6E-04	2.6E-03	3.1E-05	4.6E-05	4.7E-07	9.8E-05	1.2E-06
		Dose per year	4.6E-06	2.6E-03	3.1E-07	4.6E-07	4.7E-09	9.8E-07	1.2E-08
		LCF	2.3E-12	1.3E-08	1.6E-13	2.3E-13	1.9E-15	4.9E-10	5.9E-15

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–28 Summary of Dose Calculation Results for Salt Powder Spill (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (mrem)***	***Population (person-rem)***	***Average Individual (mrem)***	***MEI (mrem)***	***Worker (mrem)***	***Population (person-rem)***	***Average Individual (mrem)***
Hot Fuel Examination Facility Salt Powder spill	0.01	Dose per event	1.2E-04	7.1E-04	8.5E-06	1.2E-05	1.1E-06	2.7E-05	3.2E-07
		Dose per year	1.2E-06	7.1E-06	8.5E-08	1.2E-07	1.1E-08	2.7E-07	3.2E-09
		LCF	6.2E-13	3.5E-09	4.3E-14	6.2E-14	4.4E-15	1.3E-10	1.6E-15

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–29 Summary of Dose Calculation Results for Cask Drop (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***
Cask drop	0.01	Dose per event	3.0E-02	1.4E-01	1.7E-03	1.6E-03	8.4E-04	3.5E-03	4.2E-05
		Dose per year	3.0E-04	1.4E-03	1.7E-05	1.6E-05	8.4E-06	3.5E-05	4.2E-07
		LCF	1.5E-10	6.9E-07	8.3E-12	8.2E-12	3.4E-12	1.7E-08	2.1E-13

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–30 Summary of Dose Calculation Results for Cask Drop (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***
Cask Drop	0.01	Dose per event	2.4E-03	1.1E-02	1.3E-04	1.3E-04	4.9E-05	2.8E-04	3.4E-06
		Dose per year	2.4E-05	1.1E-04	1.3E-06	1.3E-06	4.9E-07	2.8E-06	3.4E-08
		LCF	1.2E-11	5.6E-08	6.7E-13	6.6E-13	2.0E-13	1.4E-09	1.7E-14

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–31 Summary of Dose Calculation Results for Single Container Transuranic Waste Fire

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***
Transuranic Waste Fire	0.001	Dose per event	5.9E-02	2.7E-01	3.3E-03	3.2E-03	2.2E-01	7.1E-03	8.5E-05
		Dose per year	5.9E-05	2.7E-04	3.3E-06	3.2E-06	2.2E-04	7.1E-06	8.5E-08
		LCF	3.0E-11	1.4E-07	1.6E-12	1.6E-12	8.7E-11	3.5E-09	4.3E-14

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–32 Summary of Dose Calculation Results for Design-Basis Seismic Event (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***
Design-Basis Seismic Event	0.0002 (Multi-facility event)	Dose per event	1.3E+01	7.0E+01	8.4E-01	9.5E-01	4.7E+00	2.8E+00	3.4E-02
		Dose per year	2.6E-03	1.4E-02	1.7E-04	1.9E-04	8.4E-04	5.6E-04	6.8E-06
		LCF	1.3E-09	7.0E-06	8.4E-11	9.5E-11	3.8E-10	2.8E-07	3.4E-12
	0.008 (HFEF)	Dose per event	12	52	0.63	0.64	4.7	1.4	0.017
		Dose per year	0.095	0.42	0.0050	0.0051	0.037	0.011	0.00013
		LCF	4.8×10^{-8}	0.00021	2.5×10^{-9}	2.6×10^{-9}	1.5×10^{-8}	5.5×10^{-6}	6.6×10^{-11}

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–33 Summary of Dose Calculation Results for Design-Basis Seismic Event (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Popula-tion (person-rem)***	***Average Individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Popula-tion (person-rem)***	***Average Individual (millirem)***
Design-Basis Seismic Event	0.0002 (Multi-facility event)	Dose per event	3.3E+00	1.5E+01	1.8E-01	1.8E-01	1.1E+01	4.0E-01	4.8E-03
		Dose per year	6.6E-04	3.0E-03	3.6E-05	3.6E-05	2.2E-03	8.0E-05	9.6E-07
		LCF	3.3E-10	1.5E-06	1.8E-11	1.8E-11	8.8E-10	4.0E-08	4.8E-13
	0.008 (HFEF)	Dose per event	3.3	14	0.17	0.18	11	0.38	0.0046
		Dose per year	0.026	0.11	0.0014	0.0014	0.088	0.0030	3.6×10^{-5}
		LCF	1.3×10^{-8}	5.7×10^{-5}	6.9×10^{-10}	7.2×10^{-10}	3.5×10^{-8}	1.5×10^{-6}	1.8×10^{-11}

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–34 Summary of Dose Calculation Results for Salt Transfer Accident (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***
Salt Transfer Accident	1.00E-07	Dose per event	1.9E-01	8.4E-01	1.0E-02	1.0E-02	7.3E-02	2.2E-02	2.6E-04
		Dose per year	1.9E-08	8.4E-08	1.0E-09	1.0E-09	7.3E-09	2.2E-09	2.6E-11
		LCF	9.5E-15	4.2E-11	5.1E-16	5.1E-16	2.9E-15	1.1E-12	1.3E-17

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–35 Summary of Dose Calculation Results for Salt Transfer Accident (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***
Salt Transfer Accident	1.00E-07	Dose per event	5.2E-02	2.4E-01	2.8E-03	2.9E-03	1.7E-01	6.2E-03	7.4E-05
		Dose per year	5.2E-09	2.4E-08	2.8E-10	2.9E-10	1.7E-08	6.2E-10	7.4E-12
		LCF	2.6E-15	1.2E-11	1.4E-16	1.4E-16	7.0E-15	3.1E-13	3.7E-18

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–36 Summary of Dose Calculation Results for Beyond-Design-Basis Seismic Event (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***
Beyond-Design-Basis Seismic Event	1.0E-05	Dose per event	2.2E+04	9.7E+04	1.2E+03	1.2E+03	3.7E+02	2.5E+03	3.1E+01
		Dose per year	2.2E-01	9.7E-01	1.2E-02	1.2E-02	3.7E-03	2.5E-02	3.1E-04
		LCF	2.2E-07	4.9E-04	5.9E-09	6.0E-09	1.5E-09	1.3E-05	1.5E-10

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–37 Summary of Dose Calculation Results for Beyond-Design-Basis Seismic Events (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***
Beyond DBE Seismic Event	1.0E-05	Dose per event	7.1E+02	3.2E+03	3.9E+01	3.8E+01	4.2E+02	8.3E+01	1.0E+00
		Dose per year	7.1E-03	3.2E-02	3.9E-04	3.8E-04	4.2E-03	8.3E-04	1.0E-05
		LCF	3.5E-09	1.6E-05	1.9E-10	1.9E-10	1.7E-09	4.1E-07	5.0E-12

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

F.2.3.3 Alternative 2 - Clean (Sodium Removal) Blanket Fuel and Package in High-Integrity Cans at ANL-W

The processing technology considered for this alternative consists of cleaning the sodium from blanket spent nuclear fuel and encasing the cleaned spent nuclear fuel in high-integrity cans. The sodium-bonded driver fuel would be processed using the electrometallurgical treatment process. The dose calculation results for this combination of processes at ANL-W facility are to be found in sections "Alternative 1 - Electrometallurgical Treatment at ANL-W" for the electrometallurgical treatment processing of the driver fuel, and "Alternative 3 - Declad/Sodium Removal at ANL-W and PUREX at Savannah River Site" for the blanket fuel. All of the electrometallurgical treatment accidents for the driver fuel are applicable to this process. For the blanket fuel, the sodium fire and the cask handling accident are applicable. The accident assumptions and parameters used in developing this information have been provided in Section 2.2 of this Appendix. EBR-II driver fuel and EBR-II blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively.

F.2.3.4 Alternative 3 - Declad/Sodium Removal at ANL-W and PUREX at SRS

The processing technology considered for this alternative consists of decladding and cleaning the sodium-bonded blanket spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W and shipment of this material to SRS for PUREX processing. In this alternative the sodium-bonded driver spent nuclear fuel is processed using the electrometallurgical treatment process at ANL-W. No driver fuel is to be shipped from ANL-W to SRS. **Tables F–38** through **F–44** provide the dose calculation results for the accidents during the PUREX process at SRS and cask drop and sodium fire accidents at ANL-W. The accident assumptions and parameters used in developing this information have been provided in Section 2.2 of this appendix. EBR-II driver fuel and EBR-II blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively.

Consequence and risk estimates are provided for both the processing of the blanket material at ANL-W prior to its shipment to SRS and for the processing of the material at SRS. Analysis results for the processing of the driver fuel can be found in the discussion of Alternative 1 "Electrometallurgical Treatment Process" and Alternative 6 (Melt and Dilute at ANL-W).

Table F–38 Summary of Dose Calculation Results for F-Canyon Fire

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
F-Canyon Fire	0.000061	Dose per event	610	36,000	2,300	5,500
		Dose per year	0.037	2.2	0.14	0.34
		LCF	1.9×10^{-8}	0.0011	5.6×10^{-8}	0.00017

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–39 Summary of Dose Calculation Results for FB-Line Explosion

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
FB-Line Explosion	0.00010	Dose per event	6.5	360	19	53
		Dose per year	0.00065	0.036	0.0019	0.0053
		LCF	3.3×10^{-10}	0.000018	7.6×10^{-10}	2.7×10^{-6}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–40 Summary of Dose Calculation Results for F-Canyon Earthquake

			95th-Percentile Meteorology		50th-Percentile Meteorology	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Worker (millirem)*	*Population (person-rem)*
F-Canyon Earthquake	0.00013	Dose per event	1,100	38,000	12,000	2,100
		Dose per year	0.14	4.9	1.6	0.27
		LCF	7.2×10^{-8}	0.0025	4.8×10^{-7}	0.00014

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–41 Summary of Dose Calculation Results for FB-Line Earthquake

			95th-Percentile Meteorology		50th-Percentile Meteorology	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Worker (millirem)*	*Population (person-rem)*
FB-Line Earthquake	0.00013	Dose per event	58	2,250	900	120
		Dose per year	0.0075	0.29	0.12	0.016
		LCF	3.8×10^{-9}	0.00015	4.7×10^{-8}	7.8×10^{-6}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–42 Summary of Dose Calculation Results for F-Canyon Criticality

			95th-Percentile Meteorology		50th-Percentile Meteorology	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Worker (millirem)*	*Population (person-rem)*
F-Canyon Criticality	0.00010	Dose per event	11	380	37	59
		Dose per year	0.0011	0.038	0.0037	0.0059
		LCF	5.5×10^{-10}	0.000019	1.5×10^{-9}	3.0×10^{-6}

MEI=- Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–43 Summary of Dose Calculation Results for ANL-W Cask Drop Accident

			95th-Percentile Meteorology		50th-Percentile Meteorology	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Worker (millirem)*	*Population (person-rem)*
Cask drop	0.01	Dose per event	2.4E-03	1.1E-02	4.9E-05	2.8E-04
		Dose per year	2.4E-05	1.1E-04	4.9E-07	2.8E-06
		LCF	1.2E-11	5.6E-08	2.0E-13	1.4E-09

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–44 Summary of Dose Calculation Results for ANL-W Sodium Fire

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
Sodium Fire during declad and clean	.008	Dose per event	5.9	26.3	0.054	0.69
		Dose per year	0.047	0.21	0.00043	0.0055
		LCF	2.4×10^{-8}	0.00011	1.7×10^{-10}	2.7×10^{-6}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

F.2.3.5 Alternative 4 - Melt and Dilute Blanket Fuel at ANL-W

The processing technology considered for this alternative consists of melting and diluting the cleaned blanket spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W. In this alternative, the sodium-bonded driver spent nuclear fuel is processed using the electrometallurgical treatment process at ANL-W. The dose calculation results for this alternative are provided elsewhere in this section. The results for the driver fuel are presented as part of the results for Alternative 1 "Electrometallurgical Treatment at ANL-W" and the results for the blanket fuel are presented as part of the results for Alternative 6 "Melt and Dilute at ANL-W," where the results for the melt and dilute processing of both driver and blanket fuel are presented. The accident assumptions and parameters used in developing this information have been provided in Section 2.2 of this Appendix. EBR-II driver and blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively.

F.2.3.6 Alternative 5 - Declad/Sodium Removal of Blanket Fuel at ANL-W, Melt and Dilute at SRS

The processing technology considered for this alternative consists of decladding, cleaning, and packaging of the blanket spent nuclear fuel at the Hot Fuel Examination facility at ANL-W and shipment of packaged blanket fuel to SRS for melt and dilute processing in the Building 105-L. In this alternative, the sodium-bonded driver spent nuclear fuel is processed either using the electrometallurgical treatment process or the melt and dilute process at ANL-W. No driver fuel is to be shipped from ANL-W to SRS. **Tables F–45** through **F–50** provide the dose calculation results for the melt and dilute process at SRS. The accident assumptions and parameters used in developing this information has been provided in Section 2.2 of this Appendix. EBR-II driver fuel and EBR-II blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively.

Consequence and risk estimates are provided for both the processing of the blanket material at ANL-W prior to its shipment to SRS, and for the processing of the material at SRS. Analysis results for the processing of the driver fuel can be found in the discussion for Alternative 1 "Electrometallurgical Treatment Process."

Table F–45 Summary of Dose Calculation Results for L-Area Waste Handling Accident

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
L-Area Waste Handling Accident	0.024	Dose per event	2.1	42	0.17	3.6
		Dose per year	0.05	1.01	0.0041	0.086
		LCF	2.6×10^{-8}	0.00050	1.6×10^{-9}	0.000043

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–46 Summary of Dose Calculation Results for L-Area Loss of Power

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
L-Area Loss of Power	0.006	Dose per event	2,100	42,000	140	3,500
		Dose per year	12.6	250	0.84	21
		LCF	6.6×10^{-6}	0.13	3.4×10^{-7}	0.011

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–47 Summary of Dose Calculation Results for L-Area Loss of Cooling Water Integrity

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
L-Area Loss of Cooling Water Integrity	0.05	Dose per event	120	2,800	1.3	500
		Dose per year	6.0	140	0.07	25
		LCF	3.0×10^{-6}	0.070	2.6×10^{-8}	0.013

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–48 Summary of Dose Calculation Results for L-Area Fire

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
L-Area Waste Handling Accident	0.075	Dose per event	86	1,700	6.3	140
		Dose per year	6.5	130	0.47	11
		LCF	3.2×10^{-6}	0.064	1.9×10^{-7}	0.0053

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–49 Summary of Dose Calculation Results for ANL-W Cask Drop Accident

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	*Population (person-rem)*
Cask drop	0.01	Dose per event	2.4E-03	1.1E-02	4.9E-05	2.8E-04
		Dose per year	2.4E-05	1.1E-04	4.9E-07	2.8E-06
		LCF	1.2E-11	5.6E-08	2.0E-13	1.4E-09

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–50 Summary of Dose Calculation Results for ANL-W Sodium Fire

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
Sodium Fire during declad and clean	.008	Dose per event	5.9	26.3	0.054	0.69
		Dose per year	0.047	0.21	0.00043	0.0055
		LCF	2.4×10^{-8}	1.1×10^{-4}	1.7×10^{-10}	2.7×10^{-6}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

F.2.3.7 Alternative 6 - Melt and Dilute at ANL-W

The processing technology considered for this alternative consists of cleaning both blanket and driver fuel and melting and diluting the fuel at the Hot Fuel Examination Facility at ANL-W. **Tables F–51** through **F–57** provide the dose calculation results for the melt and dilute process at the ANL-W. The accident assumptions and parameters used in developing this information has been provided in Section 2.2 of this appendix. EBR-II driver fuel and EBR-II blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively.

Consequence and risk estimates are provided for both the declad and clean processing and the melt and dilute processing of the sodium-bonded spent nuclear fuel.

Table F–51 Summary of Dose Calculation Results for Melt and Dilute DBE (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Design-Basis Earthquake includes sodium fire	0.008	Dose per event	19,000	89,400	1,080	1,080	838	2,250	27
		Dose per year	152	715.2	8.64	8.64	6.7	18	0.216
		LCF	7.6×10^{-5}	0.36	4.3×10^{-6}	4.3×10^{-6}	2.7×10^{-6}	0.009	1.1×10^{-7}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–52 Summary of Dose Calculation Results for Melt and Dilute DBE (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (millirem)***	***Worker (millirem)***	***Population (person-rem)***	***Average individual (millirem)***
Design-Basis Earthquake includes sodium fire	0.008	Dose per event	471	2240	26.9	27	15.2	56.1	.676
		Dose per year	3.77e-00	17.92	2.15e-01	2.16e-01	1.22e-01	0.4488	5.41e-03
		LCF	1.88e-06	8.96e-03	1.08e-07	1.08e-07	4.86e-08	2.24e-04	2.70e-09

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–53 Summary of Dose Calculation Results for Melt and Dilute Waste Handling Accident (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (millirem)***	***Worker (millirem)***	***Population (person-rem)***	***Average individual (millirem)***
Waste Handling Accident	0.024	Dose per event	597	2820	34	33.9	26.7	70.8	.852
		Dose per year	14.33	67.68	8.16e-01	8.14e-01	6.41e-01	1.6992	2.04e-02
		LCF	7.16×10^{-6}	3.38e-02	4.08e-07	4.07e-07	2.56e-07	0.00085	1.02e-08

MEI - Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–54 Summary of Dose Calculation for Melt and Dilute Waste Handing Accident (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (milli-rem)***	***Worker (millirem)***	***Population (person-rem)***	***Average individual (millirem)***
Waste Handling Accident	0.024	Dose per event	14.9	70.8	0.852	0.853	0.489	1.77	0.0214
		Dose per year	0.358	1.70	0.0204	0.0205	0.0117	0.0425	0.000514
		LCF	1.8×10^{-7}	0.00085	1.0×10^{-8}	1.0×10^{-8}	4.7×10^{-9}	0.000021	2.6×10^{-10}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–55 Summary of Dose Calculation Results for Melt and Dilute Criticality Accident (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (milli-rem)***	***Worker (millirem)***	***Population (person-rem)***	***Average individual (millirem)***
Criticality	0.0003	Dose per event	.516	1.6	.0192	.0832	.467	.0854	1.03e-6
		Dose per year	1.55e-04	0.00048	0.000006	2.50e-05	1.40e-04	2.56e-05	3.09e-10
		LCF	7.74e-11	2.40e-07	2.88e-12	1.25e-11	5.60e-11	1.28e-08	1.55e-16

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–56 Summary of Dose Calculation Results for Melt and Dilute Na Fire (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (millirem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average individual (millirem)***
Sodium fire	0.008	Dose per event	282	1,260	15.2	15.6	2.59	33	.397
		Dose per year	2.26e-00	10.08	0.1216	0.1248	0.02072	0.264	0.003176
		LCF	1.13e-06	5.04e-03	6.08e-08	6.24e-08	8.29e-06	1.32e-07	1.59e-09

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–57 Summary of Dose Calculation Results for Melt and Dilute Na Fire (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average individual (millirem)***
Sodium fire	0.008	Dose per event	5.9	26.3	.317	.326	.0541	.689	.0083
		Dose per year	4.72e-02	0.2104	2.54e-03	2.61e-03	4.33e-04	0.005512	6.64e-05
		LCF	2.36e-08	1.05e-04	1.27e-09	1.30e-09	1.73e-10	2.76e-06	3.32e-11

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

F.3 IMPACT OF HAZARDOUS CHEMICAL ACCIDENTS TO HUMAN HEALTH

F.3.1 Chemical Accident Analysis Methodology

Factors such as receptor locations, terrain, meteorological conditions, release conditions, and characteristics of the chemical inventory are required as input parameters for hand calculations or computer codes to determine human exposure from airborne releases of toxic chemicals. This section gives a general narrative about these input parameters with degrees of conservatism noted, and describes the computer models used to perform exposure estimates. EPIcode™ is the computer code chosen for estimating airborne concentrations resulting from most releases of toxic chemicals (Homann 1988).

F.3.1.1 *EPIcode™*

EPIcode™ uses the well-established Gaussian Plume Model to calculate the airborne toxic chemical concentrations at the receptor locations. The EPIcode™ library contains information on over 600 toxic substances listed in the Threshold Limit Values for Chemical Substances and Physical Agents and Biomedical Exposure Indices (ACGIH 1994). The types of releases that can be modeled, and associated input parameters, are discussed below.

Continuous release models require specifying the source term as an ambient concentration and a release rate. For term releases, the user specifies the release duration and the total quantity of material released. Area continuous and area term releases are useful in calculating the effects of a release from pools of spilled volatile liquids. The user must enter the effective radius of the release; e.g., the radius of the circle encompassing the spill area. (Also entered is the temperature of the pool and ambient temperature to establish release rate from a liquid spill.) An upwind virtual point source, which results in an initial lateral diffusion equal to the effective radius of the area source, is used to model an area release.

By specifying a release quantity, release duration and release area, the user effectively proposes a release rate per unit spill area. EPIcode™ confirms that the volatility of the spilled substance can support such a release rate. If the proposed release rate exceeds the saturation conditions at the release temperature, the EPIcode™ calculates a lower release rate and a corresponding longer release time.

In calculating effective release height, the actual plume height may not be the physical release height, e.g., the stack height. Plume rise can occur because of the velocity of a stack emission and the temperature differential between the stack effluent and the surrounding air. EPIcode™ calculates both the momentum plume rise and the buoyant plume rise and chooses the greater of the two results.

Concentrations of chemical and radiological materials is highly dependent upon the effective release height (i.e, the effective height of a stack or an evaporating pool of spilled material). Thermal buoyancy was taken into consideration for those scenarios involving fire or heat source. In those cases, a temperature of 200 °C (392 °F) was assumed for the thermal buoyancy term. This is conservative, since expected surface temperatures and resulting buoyancy terms are expected to be greater in actual fires or heat sources.

In this application, the standard terrain calculation of EPIcode™ is always used. Except as otherwise noted, both the 50th and 95th percent meteorological (stability class and wind speed) conditions for INEEL are input into EPIcode™. The receptor height is always ground level (0 meters) and the mixing layer height is always 400 meters (1,300 feet).

As described in its user manual (Homann 1988), the EPIcode™ also performs the following steps:

- Treats a release as instantaneous versus continuous depending upon the plume length at the specific downwind location being considered
- Corrects the concentration for sampling time
- Adjusts the wind speed for release height
- Depletes the plume as a function of downwind distance
- Adjusts the standard deviations of the crosswind and vertical concentrations for brief releases.

As output, EPIcode™ can generate data plots of mean toxic chemical concentration (during a specified averaging time) as a function of downwind distance. From these graphs and numerical output, the concentrations at receptor locations are determined and evaluated for health effects.

EPIcode™ was selected as the computer code for release analysis of chemicals amenable to Gaussian modeling after comparison with a number of codes, primarily CHARM and ARCHIE. It was judged easier to use for this simple application than either the more sophisticated, proprietary CHARM code or the comparable, public domain ARCHIE code. The SLAB code had previously been selected by INEEL as the most appropriate of the refined dispersion models (such as CHARM) for modeling special case releases, such a dense gas dispersion, where negative buoyancy effects must be considered. However, because chemical accident scenarios involving dispersion of denser-than-air gases were not considered in this analysis, the SLAB model was not used. EPIcode™ was judged to be a satisfactory code for the inventory of chemicals analyzed.

F.3.1.2 Health Effects

Hazardous constituents dispersed during an accident could induce adverse health effects among exposed individuals. This possible impact is assessed by comparing the airborne concentrations of each substance at specified downwind receptor locations to standard accident exposure guidelines for chemical toxicity.

Where available, the Emergency Response Planning Guideline values are used for this comparison. The guideline values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects. The Emergency Response Planning Guideline values are specific for each substance, and are derived for each of three general severity levels:

- *ERPG-1:* The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.

- *ERPG-2:* The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.

- *ERPG-3:* The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing or developing life-threatening health effects.

Where ERPG values have not been derived for a toxic substance, other chemical toxicity values are substituted, as follows:

- For ERPG-1, threshold limit value/time-weighted average (TLV-TWA) values (ACGIH 1994) are substituted: The TWA is the time-weighted average concentration for a normal 8-hour workday and a 40-hour workweek, to which nearly all workers may be repeatedly exposed, day after day, without adverse effect.

- For ERPG-2, level of concern (LOC) values (equal to 0.1 of IDLH—see below) are substituted: LOC is defined as the concentration of a hazardous substance in air, above which there may be serious irreversible health effects or death as a result of a single exposure for a relatively short period of time (EPA 1987).

- For ERPG-3, immediately dangerous to life or health (IDLH) values are substituted: IDLH is defined as the maximum concentration from which a person could escape within 30 minutes without a respirator and without experiencing any escape impairing or irreversible side effects (HHS 1997).

Possible health effects associated with exceeding an ERPG-2 or -3 are specific for each substance of concern, and must be characterized in that context. When concentrations are found to exceed an ERPG or substitute value, specific toxicological effects for the chemicals of concern are considered in describing possible health effects associated with exceeding a threshold value.

ERPG values are based upon a one-hour exposure of a member of the general population. In this analysis, ERPG values are applied only to time-averaged exposures of one hour or less in duration. This approach provides an additional element of conservatism in the evaluation of accidents with releases that are significantly less than one hour. In instances of very short exposures to substances whose effects are concentration-dependent (i.e., chlorine) and where toxicological data support analysis at short exposure times, threshold concentrations of lethality are reported (the minimum concentration necessary to cause a fatality).

F.3.2 Accident Scenario Selection and Descriptions

Toxic Chemical Accidents at ANL-W

This section describes the nonradiological consequences of the abnormal event associated with handling uranium ingots. Four accidents have been identified at ANL-W that have the potential to result in the release of either uranium or uranium and cadmium. These accidents, a uranium handling accident, a design-basis uranium fire, a design-basis seismic event, and a beyond-design-basis seismic event are discussed below.

Uranium Handling Accident

Uranium ingots with a 20 percent enrichment or less, from the electrometallurgical treatment process are transferred from the Fuel Conditioning Facility to storage at the Fuel Assembly Storage Building (ANLW-787) or the ZPPR Material Building (ANLW-792). Transfers are made using a forklift or by truck. The uranium ingots weigh about 6 kilograms (13 pounds) each. They are stored in containers holding about 140 kilograms

(310 pounds) of ingots. Depleted uranium is also stored at ANL-W in containers holding 1,350 kilograms (3,000 pounds) of ingots.

The accident involves a handling accident in which an ingot of uranium is dropped onto a hard surface, small particles are broken off the ingot, and the pyrophoric properties of the uranium result in ignition of the particles. The resulting small fire is assumed to consume 10 percent of the ingot. The accident could occur as a result of a container drop during handling, a drop during inspection, or due to a seismic event. The release occurs at ground level. A handling accident resulting in the drop of a uranium ingot may be anticipated to occur over the life of the project (0.1). The conditional probability of a fire that consumes 10 percent of the dropped ingot is assumed to be 1 in 10 drops at most. The estimated frequency of the accident is therefore 0.01.

The material at risk is one 6 kilograms ingot of uranium. The damage ratio is 0.1, as it is assumed that 10 percent of the ingot is consumed in the fire. The airborne release fraction is 0.0001, and the respirable fraction is 1.0 for metal fires (DOE 1994b). The accident is assumed to occur outdoors or with little confinement. A leak path factor of 1.0 is assumed. This information is summarized in **Table F–58**.

Table F–58 Toxic Chemical Source Term for Uranium Handling Accident

Chemical	*MAR (grams)*	*Damage Ratio*	*ARF*	*RF*	*LPF*	*Released Grams*
Uranium	6.00E+03	0.1	1.00E-04	1.00E+00	1	6.00E-02

MAR=material at risk, ARF=airborne release fraction, RF=respirable fraction, LPF=leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Accident: Design-Basis Uranium Fire

Uranium ingots with a 20 percent enrichment, or less, from the electrometallurgical treatment process are transferred from the Fuel Conditioning Facility to storage at the Fuel Assembly Storage Building (ANLW-787) or the ZPPR Material Building (ANLW-792). Transfers are made using a forklift or by truck. The uranium ingots weigh about 6 kilograms (13 pounds) each. They are stored in containers holding about 140 kilograms (310 pounds) of ingots. Depleted uranium is also stored at ANL-W in containers holding 1,350 kilograms (3,000 pounds) of ingots.

The accident involves a fire consuming the equivalent of one container of uranium (140 kilograms). The accident could occur due to a handling accident, poor housekeeping in the storage area, electrical failure, or seismic event. The uranium is in the form of ingots that have a small surface area to mass ratio. Uranium is stored in metal containers that are not combustible. The postulated accident is estimated to have a frequency of 1×10^{-5} per year (see the discussion of radiological accidents in section F.2).

The material at risk is one 140 kilograms container of uranium. The damage ratio is 1.0, as it is assumed that all of the uranium is consumed in the fire. The airborne release fraction is 0.0001, and the respirable fraction is 1.0 for metal fires (DOE 1994b). The accident is assumed to occur outdoors or with little confinement (i.e., door to the storage facility open). A leak path factor of 1.0 is assumed. This information is summarized in **Table F–59**.

Table F–59 Toxic Chemical Source Term for Uranium Fire

Chemical	*MAR (grams)*	*Damage Ratio*	*ARF*	*RF*	*LPF*	*Released Grams*
Uranium	1.40E+04	1	1.00E-04	1.00E+00	1	1.40E+00

MAR = material at risk, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Design-Basis Seismic Event - Multi-facility Effects

This event is the same event as described under radiological accidents for the electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W. The material at risk and release fraction information is summarized in **Table F-60**.

Table F–60 Toxic Chemical Source Term for Design-Basis Seismic Event

Chemical	*MAR (grams)*	*Damage Ratio*	*ARF/RF*	*LPF*	*Released Grams*
Uranium	17	1	2.5×10^{-6}	1	0.000043

MAR = material at risk, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

Beyond-Design-Basis Seismic Event – Multi-facility Effects

This event is the same event as described under the radiological accidents for the electrometallurgical treatment at ANL-W. The ARF/RF for cadmium is 2.5×10^{-6} (Slaughterbeek et al 1995). The material at risk and release fraction information is summarized in **Table F–61**.

Table F–61 Toxic Chemical Source Term for Beyond Design-Basis Seismic Event

Chemical	*MAR (kilograms)*	*Damage Ratio*	*ARF/RF*	*LPF*	*Released Kilograms*
Cadmium	1,000	1	2.5×10^{-6}	1	0.0025
Uranium	17	1	2.5×10^{-6}	1	0.000043

MAR = material at risk, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

Liquid Sodium Fire

This event is the event described under radiological accidents for the melt and dilute processing at ANL-W. The accident is associated with the fuel cleaning process used during the melt and dilute process or in preparation of the fuel for shipment to SRS for processing.

The accident involves a fire during the declad and clean processing of the spent nuclear fuel due to a breach of the Hot Fuel Examination Facility and exposure of liquid sodium to the air. The most probable cause of air inleakage is expected to be a seismic event. As discussed in the radiological accident description, this event has been assumed to occur with a frequency of 0.008 per year. The sodium at risk is the material cleaned from the spent nuclear fuel and is conservatively estimated to be half of all of the sodium contained in spent nuclear fuel, 300 kilograms. The release fraction information is provided in **Table F–62**. The assumption that all of the sodium is converted to sodium hydroxide and volatilized by the fire results in the airborne release fraction/respirable fraction value of 1.0.

Table F–62 Toxic Chemical Source Term for Sodium Fire in the Hot Fuel Examination Facility

Chemical	*MAR (kilograms)*	*Damage Ratio*	*ARF/RF*	*LPF*	*Released Kilograms*
Sodium	330	1	1	0.125	41.3

MAR = material at risk, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

Savannah River Site

The SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998) analyzed the consequences of accidental releases of hazardous chemicals for operations located in F-Area. These accidents involved the spill of materials associated with the wet storage of spent nuclear fuel in the F-Area. The activities associated with processing the sodium-bonded spent nuclear fuel are not expected to result in the introduction of additional hazardous materials or additional accident scenarios. Therefore, the accident scenarios identified in the SRS Spent Nuclear Fuel Management Draft EIS have been selected as representing the hazardous chemical accidents associated with processing of sodium-bonded spent nuclear fuel.

F.3.3 Accident Analyses Consequences and Risk Results

Tables F–63 through **F–67** provide the chemical risk calculation results for the electrometallurgical treatment process related accidents at the ANL-W facility. **Table F–68** reproduces the consequences from hazardous chemical accidents at SRS, as originally developed for the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998).

Table F–63 Summary of Toxic Chemical Exposure Results for Handling Accident

Receptor Location	*Chemical*	*Concentration (mg/m^3)*	*Fraction of ERPG*	*ERPG-1 value*
Noninvolved worker at 100 meters	Uranium	1.77E-04	2.95E-04	0.6 mg/m^3
Maximally Exposed Offsite Individual	Uranium	1.14E-08	1.9E-08	0.6 mg/m^3

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–64 Summary of Toxic Chemical Exposure Results for Uranium Fire

Receptor Location	*Chemical*	*Concentration (mg/m^3)*	*Fraction of ERPG*	*ERPG-1 value*
Noninvolved worker at 100 meters	Uranium	4.13E-03	6.88E-03	0.6 mg/m^3
Maximally Exposed Offsite Individual	Uranium	2.65E-07	1.08E-07	0.6 mg/m^3

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–65 Summary of Toxic Chemical Exposure Results for Design-Basis Seismic Event

Receptor Location	*Chemical*	*Concentration (mg/m^3)*	*Fraction of ERPG*	*ERPG-1 value*
Noninvolved worker at 100 meters (max is at 230 meters)	Uranium	100m 1.29E-10 230m 1.03E-09	100m 2.15E-10 230m 1.72E-09	0.6mg/m^3
Maximally Exposed Offsite Individual	Uranium	4.25E-11	7.08E-11	0.6 mg/m^3

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–66 Summary of Toxic Chemical Exposure Results for Beyond-Design-Basis Seismic Event

Receptor Location	*Chemical*	*Concentration (mg/m^3)*	*Fraction of ERPG*	*ERPG-1 value*
Noninvolved worker at 100 meters	Cadmium	7.5E-06	2.5E-04	0.03 mg/m^3
	Uranium	1.27E-07	2.12E-07	0.6 mg/m^3
Maximally Exposed Offsite Individual	Cadmium	4.58E-10	1.53E-08	0.03 mg/m^3
	Uranium	8.15E-12	1.36E-11	0.6 mg/m^3

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–67 Summary of Toxic Chemical Exposure for Sodium Fire

Receptor Location	*Chemical*	*Concentration (mg/m^3)*	*Fraction of PEL-TWA*	*Fire PEL-TWA*
Noninvolved worker at 100 meters	Sodium hydroxide	0.15	0.075	2 mg/m^3
Maximally Exposed Offsite Individual	Sodium hydroxide	0.002	0.001	2 mg/m^3

Table F–68 Summary of Toxic Chemical Exposure Results for Wet Storage Container Ruptures at SRS

Frequency (event/year)	*Receptor*	*Chemical*	*Concentration*[a]	*Fraction of PEL-TWA*	*PEL-TWA*
0.005	Noninvolved Worker	Sodium hydroxide	less than PEL-TWA	N/A[1]	2 mg/m^3
0.005	Noninvolved worker at 640 meters	Nitric acid	3.1×10^{-3} mg/m^3	0.00062	5 mg/m^3
	Maximally Exposed Offsite Individual		4.0×10^{-4} mg/m^3	0.00008	5 mg/m^3
0.005	Noninvolved Worker	Sodium nitrite	6.0×10^{-3} mg/m^3	0.0012[2]	2 mg/m^3 [2]

[a] SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998)
[1] Not available – SRS Spent Nuclear Fuel EIS states only that concentration in less than lowest PEL-TWA.
[2] No PEL-TWA for this specific chemical. Lowest PEL-TWA of potential chemical reaction products is 2 mg/m^3.

Table F-69 provides a summary of the applicability of the analyzed toxic chemical accidents to each of the alternatives considered in detail for the processing of the sodium-bonded spent nuclear fuel. The hazardous chemical accidents applicable to the No Action Alternative include only those accidents associated with operation at ANL-W. Additionally, only three of the four accidents identified, excluding the uranium fire, can be associated with this alternative. Accidents associated with this alternative are the result of activities from the final processing of the sodium-bonded spent nuclear fuel treated with the electrometallurgical treatment process as part of the Electrometallurgical Treatment Demonstration Program. Alternatives 2 through 5 include electrometallurgical treatment of at least some of the sodium-bonded spent nuclear fuel and decladding and cleaning of blanket fuel, therefore, all of the identified toxic chemical accidents at ANL-W are applicable to these alternatives. Alternative 1 includes electrometallurgical treatment of a fuel but no declad and clean operations, therefore for this alternative all ANL-W accidents except the sodium fire are applicable. Processing of the spent nuclear fuel at SRS occurs only in Alternatives 3 and 5, and the accidents at SRS are applicable to these two alternatives. The accidents identified for ANL-W are associated with the electrometallurgical treatment of the sodium-bonded spent nuclear fuel. Alternative 6 does not include this treatment option and no other accidental releases of hazardous chemical were identified.

Table F–69 Applicability of Hazardous (Toxic) Chemical Accidents to Sodium-Bonded Spent Nuclear Fuel Processing Alternatives

Alternative		*ANL-W Toxic Chemical Accidents*	*SRS Toxic Chemical Accidents*
	No Action	Uranium handling accident Uranium fire Design-basis seismic event	Not Applicable
1	Electrometallurgical Treatment	Uranium handling accident Uranium fire Design-basis seismic event Beyond-design-basis seismic event	Not Applicable
2	High-Integrity Cans (blanket), electrometallurgical treatment (driver)	Alternative 1 accidents plus sodium fire	Not applicable
3	Declad and Clean at ANL-W and PUREX at SRS (blanket), electrometallurgical (driver)	Alternative 1 accidents plus sodium fire	Wet storage, container rupture
4	Melt and Dilute at ANL-W (blanket) electrometallurgical (driver)	Alternative 1 accidents plus sodium fire	Not applicable
5	Melt and Dilute at SRS (blanket), electrometallurgical (driver)	Alternative 1 accidents plus sodium fire	Wet storage, container rupture
6	Melt and Dilute at ANL-W (blanket and driver)	Sodium fire	Not applicable

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APPENDIX G
EVALUATION OF HUMAN HEALTH EFFECTS OF OVERLAND TRANSPORTATION

G.1 INTRODUCTION

Overland transportation of any commodity involves a risk to both transportation crew members and members of the public. This risk results directly from transportation-related accidents and indirectly from the increased levels of pollution from vehicle emissions, regardless of the cargo. The transportation of certain materials, such as hazardous or radioactive waste, can pose an additional risk due to the unique nature of the material itself. To permit a complete appraisal of the environmental impacts of the proposed action and alternatives, the human health risks associated with the overland transportation of spent nuclear fuel are assessed.

This appendix provides an overview of the approach used to assess the human health risks that may result from overland transportation. The topics in this appendix include the scope of the assessment, packaging and determination of potential transportation routes, analytical methods used for the risk assessment (e.g., computer models), and important assessment assumptions. It also presents the results of the assessment. In addition, to aid in the understanding and interpretation of the results, specific areas of uncertainty are described with an emphasis on how the uncertainties may affect comparisons of the alternatives.

The risk assessment results are presented in this appendix in terms of "per-shipment" risk factors, as well as for the total risks for a given alternative. Per-shipment risk factors provide an estimate of the risk from a single shipment. The total risks for a given alternative are found by multiplying the expected number of shipments by the appropriate per-shipment risk factors.

G.2 SCOPE OF ASSESSMENT

The scope of the overland transportation human health risk assessment, including the alternatives and options, transportation activities, potential radiological and nonradiological impacts, and transportation modes considered, is described below. Additional details of the assessment are provided in the remaining sections of the appendix.

Proposed Action and Alternatives

The transportation risk assessment conducted for this environmental impact statement (EIS) estimates the human health risks associated with the transportation of sodium-bonded spent nuclear fuel for the 6 alternatives. There are 6 different shipping arrangements for 8 fuel types that cover the 6 alternatives evaluated. Consistent with the scope of the overland transportation human health risks, this evaluation focuses on using onsite and offsite public highways.

Transportation-Related Activities

The transportation risk assessment is limited to estimating the human health risks incurred during overland transportation for each alternative. The risks to workers or to the public during loading, unloading, and handling prior to or after shipment are not included in the overland transportation assessment, but are addressed in Appendix F of this EIS. The transportation risk assessment does not address possible impacts from increased transportation levels on local traffic flow, noise levels, or infrastructure.

Radiological Impacts

For each alternative, radiological risks (i.e., those risks that result from the radioactive nature of the spent nuclear fuel) are assessed for both incident-free (i.e., normal) and accident transportation conditions. The radiological risk associated with incident-free transportation conditions would result from the potential exposure of people to external radiation in the vicinity of a loaded shipment. The radiological risk from transportation accidents would come from the potential release and dispersal of radioactive material into the environment during an accident and the subsequent exposure of people.

All radiological impacts are calculated in terms of committed dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent (see 10 CFR 20), which is the sum of the effective dose equivalent from external radiation exposure and the 50-year committed effective dose equivalent from internal radiation exposure. Radiation doses are presented in units of roentgen equivalent man (rem) for individuals and person-rem for collective populations. The impacts are further expressed as health risks in terms of latent cancer fatalities and cancer incidence in exposed populations using the dose-to-risk conversion factors established by the National Council on Radiation Protection and Measurement (NCRP 1993).

Nonradiological Impacts

In addition to the radiological risks posed by overland transportation activities, vehicle-related risks are also assessed for nonradiological causes (i.e., causes related to the transport vehicles and not the radioactive cargo) for the same transportation routes. The nonradiological transportation risks, which would be incurred for similar shipments of any commodity, are assessed for both incident-free and accident conditions. The nonradiological risks during incident-free transportation conditions would be caused by potential exposure to increased vehicle exhaust emissions. The nonradiological accident risk refers to the potential occurrence of transportation accidents that directly result in fatalities unrelated to the shipment of cargo. State-specific transportation fatality rates are used in the assessment. Nonradiological risks are presented in terms of estimated fatalities.

Transportation Modes

All shipments are assumed to take place by truck transportation modes.

Receptors

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. Potential risks are estimated for the collective populations of exposed people and for the hypothetical maximally exposed individual. For incident-free operation, 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 would be an individual located 33 meters (105 feet) directly downwind from the accident. The collective population risk is a measure of the radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing various alternatives.

G.3 Packaging and Representative Shipment Configurations

Regulations that govern the transportation of radioactive materials are designed to protect the public from the potential loss or dispersal of radioactive materials, as well as from routine radiation doses during transit. The

primary regulatory approach to promote safety is the specification of standards for the packaging of radioactive materials. Because packaging represents the primary barrier between the radioactive material being transported and radiation exposure to the public and the environment, packaging requirements are an important consideration for transportation risk assessment. Regulatory packaging requirements are discussed briefly below and in Chapter 4. The representative packaging and shipment configurations assumed for this EIS also are described below.

G.3.1 Packaging Overview

Although several Federal and state organizations are involved in the regulation of radioactive waste transportation, primary regulatory responsibility resides with the U.S. Department of Transportation and the U.S. Nuclear Regulatory Commission (NRC). All transportation activities must take place in accordance with the applicable regulations of these agencies as specified in 49 CFR 172 and 173 and 10 CFR 71.

Transportation packaging for small quantities of radioactive materials must be designed, constructed, and maintained to contain and shield their contents during normal transport conditions. For large quantities and for more highly radioactive material, such as high-level radioactive waste or spent nuclear fuel, they must contain and shield their contents in the event of severe accident conditions. The type of packaging used is determined by the total radioactive hazard presented by the material within the packaging. Four basic types of packaging are used: Excepted, Industrial, Type A, and Type B. Another packaging option, "Strong, Tight," is still available for some domestic shipments.

Excepted packages are limited to transporting materials with extremely low-levels of radioactivity. Industrial packages are used to transport materials that, because of their low concentration of radioactive materials, present a limited hazard to the public and the environment. Type A packages are designed to protect and retain their contents under normal transport conditions and must maintain sufficient shielding to limit radiation exposure to handling personnel. These packages are used to transport radioactive materials with higher concentrations or amounts of radioactivity than Excepted, or Industrial packages. Strong, Tight packages are used in the United States for shipment of certain materials with low-levels of radioactivity, such as natural uranium and rubble from the decommissioning of nuclear reactors. Type B packages are used to transport material with the highest radioactivity levels, are designed to protect and retain their contents under transportation accident conditions, and are described in more detail in the following sections.

G.3.2 Regulations Applicable to Type B Casks

Regulations for the transport of radioactive materials in the United States are issued by the U.S. Department of Transportation and are codified in 49 CFR 173. The regulation authority for radioactive materials transport is jointly shared by the U.S. Department of Transportation and the NRC. As outlined in a 1979 Memorandum of Understanding with the NRC, the U.S. Department of Transportation specifically regulates the carriers of spent nuclear fuel and the conditions of transport, such as routing, handling and storage, and vehicle and driver requirements. The U.S. Department of Transportation also regulates the labeling, classification, and marking of all spent nuclear fuel packages. The NRC regulates the packaging and transport of spent nuclear fuel for its licensees, which include commercial shippers of spent nuclear fuel. In addition, NRC sets the standards for packages containing fissile materials and spent nuclear fuel.

DOE policy requires compliance with applicable Federal regulations regarding domestic shipments of spent nuclear fuel. Accordingly, DOE has adopted the requirements of 10 CFR 71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions," and 49 CFR 173, "Shippers--General Requirements for Shipping and Packaging." DOE Headquarters can issue a certificate of compliance for a package to be used only by DOE and its contractors.

G.3.2.1 Cask Design Regulations

Spent nuclear fuel is transported in robust "Type B" transportation casks that are certified for transporting radioactive materials. Casks designed and certified for spent nuclear fuel transportation within the United States must meet the applicable requirements of NRC for design, fabrication, operation, and maintenance as contained in 10 CFR 71.

Cask design and fabrication can only be done by approved vendors with established quality assurance programs (10 CFR 71.101). Cask and component suppliers or vendors are required to obtain and maintain documents that prove the materials, processes, tests, instrumentation, measurements, final dimensions, and cask operating characteristics meet the design-basis established in the Safety Analysis Report for Packaging (described in the next section) for the cask and that the cask will function as designed.

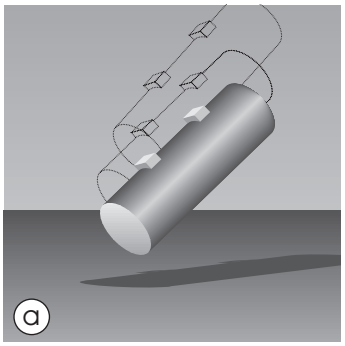
Regardless of where a transportation cask is designed, fabricated, or certified for use, it must meet certain minimum performance requirements (10 CFR 71.71–71.77). The primary function of a transportation cask is to provide containment and shielding. Regulations require that casks must be operated, inspected, and maintained to high standards to ensure their ability to contain their contents in the event of a transportation accident (10 CFR 71.87). There are no documented cases of a release of radioactive materials from spent nuclear fuel shipments, even though thousands of shipments have been made by road, rail, and water transport. Further, a number of obsolete casks have been tested under severe accident conditions to demonstrate their adherence to design criteria without failure.

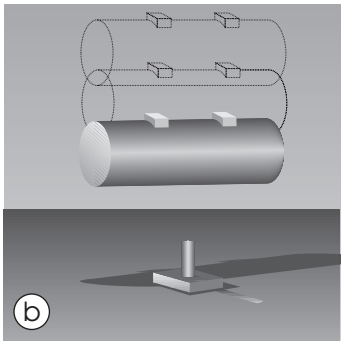
Transportation casks are built out of heavy, durable structural materials such as stainless steel. These materials must ensure cask performance under a wide range of temperatures (10 CFR 71.43). In addition to the structural materials, shielding is provided to limit radiation levels at the surface and at prescribed distances from the surface of transportation casks (10 CFR 71.47). Shielding typically consists of dense material such as lead or depleted uranium. The assemblies are supported by internal structures, called baskets, that provide shock and vibration resistance and establish minimum spacing and heat transfer to maintain the temperature of the contents within the limits specified in the Safety Analysis Report for Packaging.

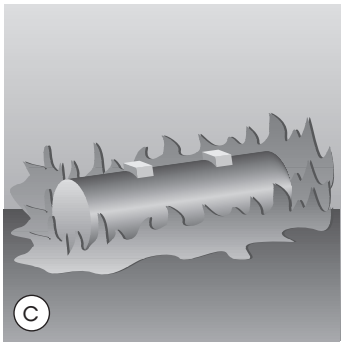
Finally, to limit impact forces and minimize damage to the structural components of a cask in the event of a transportation accident, impact-absorbing structures may be attached to the exterior of the cask. These are usually composed of balsa wood, foam, or aluminum honeycomb that is designed to readily deform upon impact to absorb impact energy. All of these components are designed to work together in order to satisfy the regulatory requirements for a cask to operate under normal conditions of transportation and maintain its integrity in an accident.

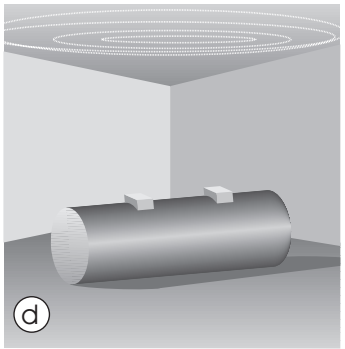
G.3.2.2 Design Certification

For certification, transportation casks must be shown by analysis and/or testing to withstand a series of hypothetical accident conditions. These conditions have been internationally accepted as simulating damage to transportation casks that could occur in most reasonably foreseeable accidents. The impact, fire, and water-immersion tests are considered in sequence to determine their cumulative effects on one package. These accident conditions are described in **Figure G–1**. The NRC issues regulations (10 CFR 71) governing the transportation of radioactive materials. In addition to the tests shown in Figure G–1, the regulations affecting Type B casks require that a transportation cask with activity greater than 10^6 curies (which is applicable to spent nuclear fuel) be designed and constructed so that its undamaged containment system would withstand an external water pressure of 2 megapascals (290 pounds per square inch), or immersion in 200 meters (656 feet) of water, for a period of not less than one hour without collapse, buckling, or allowing water to leak into the cask.









Standards for Type B Casks

For certification by the U.S. Nuclear Regulatory Commission, a cask must be shown by test or analysis to withstand a series of accident conditions without releasing its contents. These conditions have been internationally accepted as simulating damage to spent nuclear fuel casks that could occur in most severe credible accidents. The impact, fire, and water-immersion tests are considered in sequence to determine their cumulative effects on one package. An undamaged containment system is subjected to a deep water-immersion test. The details of the tests are as follows:

Impact

Free Drop (a) – The cask drops 9 meters (30 feet) onto a flat, horizontal, unyielding surface so that it strikes at its weakest point.

Puncture (b) – The cask drops 1 meter (40 inches) onto a 15.2-centimeter (6-inch) diameter steel bar at least 20.3 centimeters (8 inches) long; the bar strikes the cask at its most vulnerable spot.

Fire (c)

After the impact tests, the cask is totally engulfed in a 802 °C (1,475 °F) thermal environment for 30 minutes.

Water Immersion (d)

The cask is completely submerged under at least 1 meter (3 feet) of water for 8 hours. Additionally, undamaged containment systems (casks) are required to withstand more rigorous immersion tests.

Figure G–1 Standards for Transportation Casks

Under the Federal certification program, a Type B packaging design must be supported by a Safety Analysis Report for Packaging, which demonstrates that the design meets Federal packaging standards. The Safety Analysis Report for Packaging must include a description of the proposed packaging in sufficient detail to identify the packaging accurately and provide the basis for evaluating its design. The Safety Analysis Report for Packaging must provide the evaluation of the structural design, materials' properties, containment boundary, shielding capabilities, and criticality control, and present the operating procedures, acceptance testing, maintenance program, and the quality assurance program to be used for design and fabrication. Upon completion of a satisfactory review of the Safety Analysis Report for Packaging to verify compliance to the regulations, a Certificate of Compliance is issued.

G.3.2.3 Transportation Regulations

To ensure that the transportation cask is properly prepared for transportation, trained technicians perform numerous inspections and tests (10 CFR 71.87). These tests are designed to ensure that the cask components are properly assembled and meet leak-tightness, thermal, radiation, and contamination limits before shipping radioactive material. The tests and inspections are clearly identified in the Safety Analysis Report for Packaging and/or the Certificate of Compliance for each cask. Casks can only be operated by registered users who conduct operations in accordance with documented and approved quality assurance programs meeting the requirements of the regulatory authorities. Records must be maintained that document proper cask operations in accordance with the quality requirements of 10 CFR 71.91. Reports of defects or accidental mishandling must be submitted to NRC. DOE will be the Shipper-of-Record for the shipments that could be sent.

External radiation from a package must be below specified limits that minimize the exposure of handling personnel and the general public. For these types of shipments, the external radiation dose rate during normal transportation conditions must be maintained below the following limits of 49 CFR 173:

- 10 millirem per hour at any point 2 meters (6.6 feet) from the vertical planes projected by the outer lateral surfaces of the transport vehicle (referred to as the regulatory limit throughout this document), and
- 2 millirem per hour in any normally occupied position in the transport vehicle

Additional restrictions apply to package surface contamination levels, but these restrictions are not important for the transportation radiological risk assessment. Current contamination standards assure that workers and public receive doses much lower than those associated with radiation emitted from the casks. For risk assessment purposes, it is important to note that all packaging of a given type is designed to meet the same performance criteria. Therefore, two different Type B designs would be expected to perform similarly during incident-free and accident transportation conditions. The specific containers selected or designed, however, will determine the total number of shipments necessary to transport a given quantity material.

G.3.2.4 Communications

Proper communication assists in ensuring safe preparation and handling of transportation casks. Communication is provided by labels, markings, placarding, shipping papers, or other documents. Labels (49 CFR 172.403) applied to the cask document the contents and the amount of radiation emanating from the cask by giving the transport index. The transport index lists the ionizing radiation level (in millirem per hour) at a distance of 1 meter (3.3 feet) from the cask surface.

In addition to the label requirements, markings (49 CFR 173.471) should be placed on the exterior of the cask to show the proper shipping name and the consignor and consignee in case the cask is separated from its original shipping documents (49 CFR 172.203). Transportation casks are required to be permanently marked with the designation "Type B," the owner's (or fabricators') name and address, the Certificate of Compliance number, and the gross weight (10 CFR 71.83).

Placards (49 CFR 172.500) are applied to the transport vehicle or freight container holding the transportation cask. The placards indicate the radioactive nature of the contents. Spent nuclear fuel, which constitutes a highway route-controlled quantity or "HRCQ," must be placarded according to 49 CFR 172.507. Placards provide the first responders to a traffic or transportation accident with initial information about the nature of the contents.

Shipping papers for the spent nuclear fuel should contain the notation "HRCQ" and have entries identifying the following: the name of the shipper, emergency response telephone number, description of contents, and the shipper's certificate, as described in 49 CFR 172 Subpart C.

In addition, drivers of motor vehicles transporting radioactive material must have been trained in accordance with the requirements of 49 CFR 172.700. The training requirements include familiarization with the regulations, emergency response information, and the communication programs required by the Occupational Safety and Health Administration. Drivers are also required to have been trained on the procedures necessary for safe operation of the vehicle used to transport the spent nuclear fuel.

G.3.3 Packages Used in the Transportation of Spent Nuclear Fuel

Two Type B casks, a formerly certified type B cask, and an NRC-certified cask would provide primary transportation services for sodium-bonded fuel where public roads are involved. A commercially available cask will be certified and used for single shipments of miscellaneous sodium-bonded fuel from Tennessee and New Mexico. One other cask for onsite fuel transfers at ANL-W which does not use public roads will be employed. It is discussed below.

The TN-FSV is a certified Type B cask that would be used for intrasite transportation, and NAC-LWT would be used for the intersite transportation. The Peach Bottom (PB-1) is a formerly certified Type B cask that would be used for some of the intrasite transportation. The NRC-certified T-3 cask would be used for shipping the Fast Flux Test Facility Driver fuel from Washington to Idaho. The NRC-license is equivalent to the Type B certification described in the earlier sections.

The TN-FSV cask is a steel and lead shielded shipping cask originally designed for high temperature gas–cooled reactor fuel elements from the Fort St. Vrain reactor. The cask is a right circular cylinder, with a balsa and redwood impact limiter at each end. The cask body is made of two concentric shells of type 304 stainless steel, welded to a bottom plate and a top closure flange. The inner shell has an inside diameter of 46-centimeters (18 inches) and is 2.8 centimeters (1.1 inches) thick, and the cavity is 505 centimeters (199 inches) long. The outer shell has an outside diameter of approximately 76 centimeters (30 inches) and is 3.8 centimeters (1.5 inches) thick. The gross package weight, including the contents, is 21,319 kilograms (47,000 pounds). **Figure G–2** shows the TN-FSV.

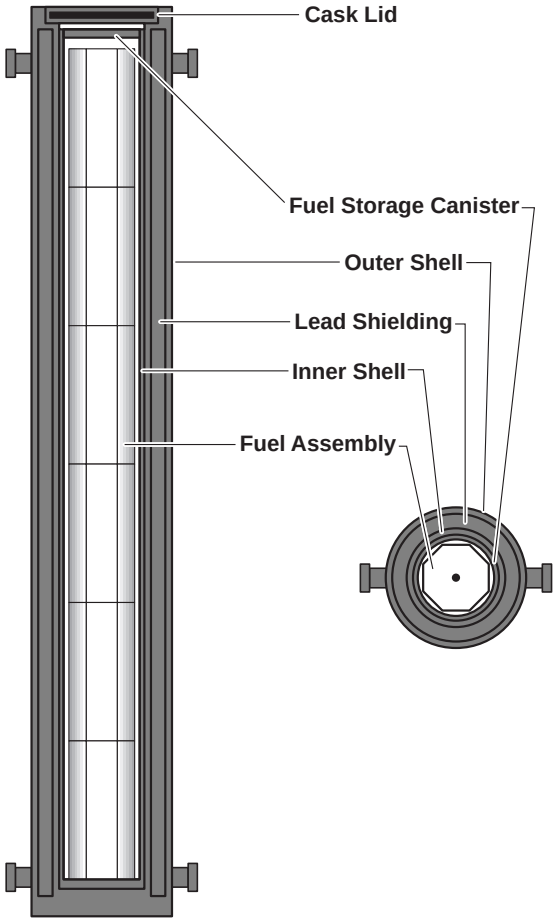


Figure G–2 TN-FSV Cask

The TN-FSV cask first received an NRC Certificate of Compliance in March 1993, and this certificate has been supplemented several times since that time. The current Certificate of Compliance would expire in May 1999, but it is likely that it would be renewed. The Certificate of Compliance would have to be supplemented for the materials that could be carried in this program. In addition to the size of the cavity, the limiting factors for this cask on the current Certificate of Compliance are a maximum of 360 watts of decay heat and a maximum total weight of contents of 2268 kilograms (5,000 pounds), including the fuel elements, fuel storage container and shield plug. (NRC 1998).

The NAC-LWT is a steel encased lead shielded shipping cask. The overall dimensions with impact limiters are 589 centimeters (232 inches) long by 165 centimeters (65 inches) in diameter. The cask body is approximately 508 centimeters (200 inches) in length and 112 centimeters (44 inches) in diameter. The cask cavity is approximately .41 cubic meters (14.5 cubic feet). The maximum weight of the package is 23,587 kilograms (52,000 pounds) and the maximum weight of the contents and basket is 1,814 kilograms (4,000 pounds). **Figure G–3** shows the NAC-LWT.

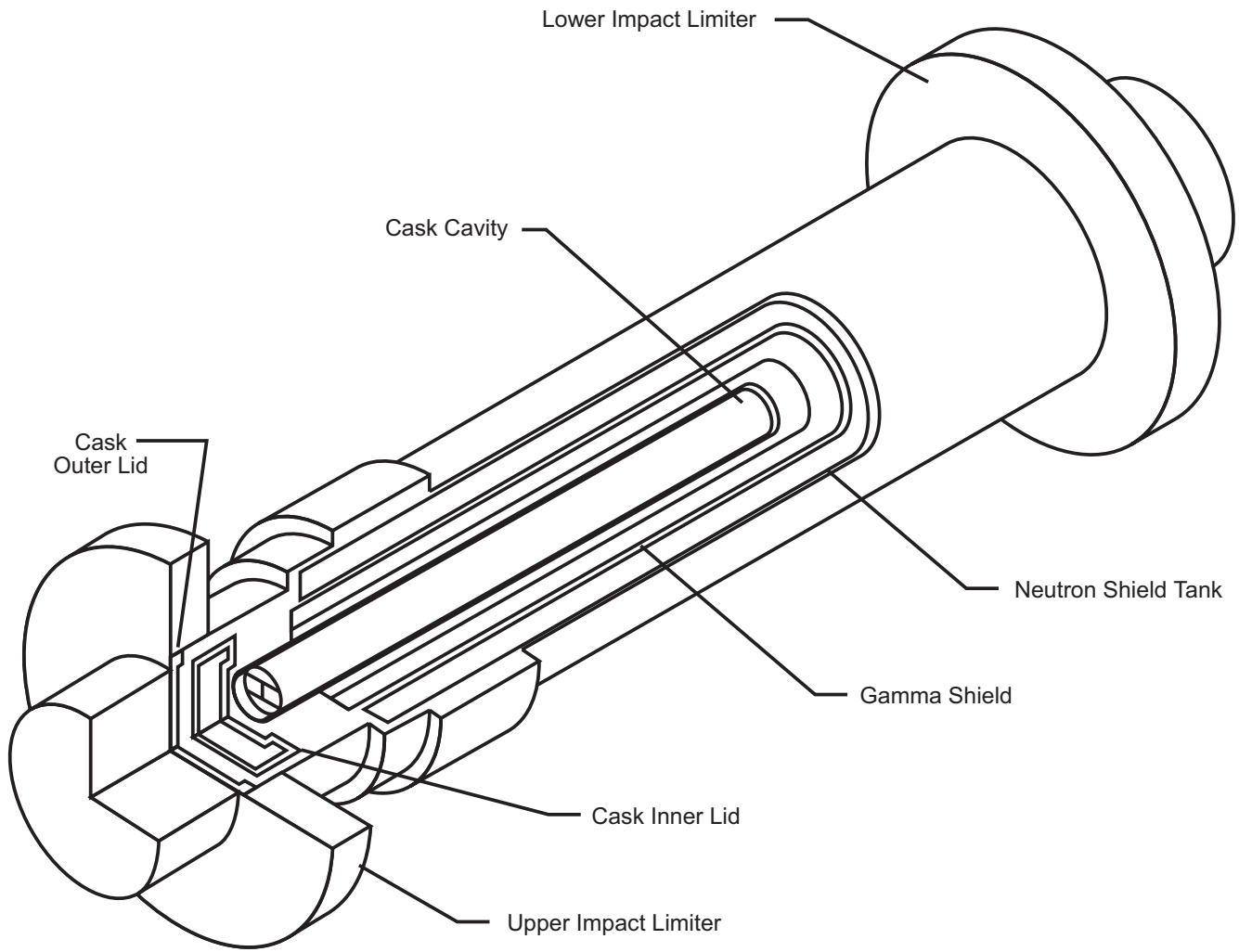


Figure G–3 Simplified Drawing of a NAC-LWT (Legal Weight Truck) Shipping Cask

The NAC-LWT first received an NRC Certificate of Compliance in March 1995, and this certificate has been supplemented several times. The current Certificate of Compliance would expire in February 2000, but it is likely that it would be renewed. The Certificate of Compliance would not need to be supplemented for the materials that could be carried in this program. The cask is designed to carry up to 42 reactor fuel assemblies. Besides the size of the cavity and weight, the limiting factor for this cask on the current Certificate of Compliance is a maximum of 210 watts of decay heat.

The intrasite transportation of Fermi-1 blanket material would use the formerly certified PB-1 cask. This cask was originally licensed for carrying Peach Bottom fuel, and was used to bring the Fermi-1 spent nuclear fuel to INTEC. The Certificate of Compliance for this cask has expired. Since the movement is a short distance on closed DOE-controlled roads, DOE procedures and NRC regulations do not require the use of a certified Type B cask. The use of formerly certified casks provides a margin of safety beyond that required by NRC regulations. The level of safety for intrasite shipments is carefully controlled by internal procedures, and the level of protection given by the PB-1 cask is approximately equivalent to that of a certified Type B cask. Since the roads are closed and site is uninhabited, there are no measurable impacts to the public.

The EBR-II driver and blanket material currently in storage at Argonne National Laboratory-West (ANL-W) is stored in HFEF-5 sealed canisters, another kind of cask. The canisters are single use, welded steel cans. DOE packs these cans in an unlicensed HFEF cask for onsite shipping. Fast Flux Test Facility driver material currently in storage at the Hanford Site would be shipped in the NRC-certified T-3 cask.

G.3.4 Ground Transportation Route Selection Process

According to DOE guidelines, spent nuclear fuel shipments must comply with both the NRC and U.S. Department of Transportation regulatory requirements. NRC regulations cover the packaging and transport of neptunium, spent nuclear fuel, whereas the U.S. Department of Transportation specifically regulates the carriers and the conditions of transport, such as routing, handling and storage, and vehicle and driver requirements. The highway routing of nuclear material is systematically determined according to U.S. Department of Transportation regulations 49 CFR 171–179 and 49 CFR 397 for commercial shipments. Specific routes cannot be publicly identified in advance for DOE's Transportation Safeguards Division's shipments because they are classified to protect national security interests.

The U.S. Department of Transportation routing regulations require that shipment of a highway route-controlled quantity of radioactive material be transported over a preferred highway network, including interstate highways, with preference toward interstate system bypasses and beltways around cities and state-designated preferred routes. A state or tribe may designate a preferred route to replace or supplement the interstate highway system in accordance with U.S. Department of Transportation guidelines (DOT 1992).

Carriers of highway route-controlled quantities are required to use the preferred network unless they are moving from their origin to the nearest interstate highway or from the interstate highway to their destination, they are making necessary repair or rest stops, or emergency conditions render the interstate highway unsafe or impassable. The primary criterion for selecting the preferred route for a shipment is travel time. Preferred routing takes into consideration accident rate, transit time, population density, activities, time of day, and day of the week.

The HIGHWAY computer code (Johnson et al. 1993) is used for selecting highway routes in the United States. The HIGHWAY database is a computerized road atlas that currently describes about 386,400 kilometers (240,000 miles) of roads. The Interstate System and all U.S. (US-designated) highways are completely described in the database. In addition, most of the principal state highways and many local and community roads are also identified. The code is updated periodically to reflect current road conditions and has been benchmarked against reported mileages and observations of commercial truck firms. Features in the

HIGHWAY code allow the user to select routes that conform to U.S. Department of Transportation regulations. Additionally, the HIGHWAY code contains data on the population densities along the routes. The distances and populations from the HIGHWAY code are part of the information used for the transportation impact analysis in this EIS.

G.4 METHODS FOR CALCULATING TRANSPORTATION RISKS

The overland transportation risk assessment method is summarized in **Figure G–4**. After the EIS alternatives were identified and the goals of the shipping campaign were understood, data was collected on material characteristics and accident parameters. Accident parameters were largely based on the NRC studies of transportation accidents undertaken for the Final Environmental Impact Statement on the Transportation of Radioactive Material by Air and Other Modes, NUREG-0170 (NRC 1977) and the Modal Study, NUREG/CR-4829 (NRC 1987).

Representative routes that may be used for the shipments were selected for risk assessment purposes using the HIGHWAY code. They do not necessarily represent the actual routes that would be used to transport nuclear materials. Specific routes cannot be identified in advance because the routes cannot be finalized until they have been reviewed and approved by the NRC. The selection of the actual route would be responsive to environmental and other conditions that would be in effect or could be predicted at the time of shipment. Such conditions could include adverse weather conditions, road conditions, bridge closures, and local traffic problems. For security reasons, details about a route would not be publicized before the shipment.

The first analytic step in the ground transportation analysis was to determine the incident-free and accident risk factors on a per-shipment basis. Risk factors, as with any risk estimate, are the product of the probability of exposure and the magnitude of the exposure. Accident risk factors were calculated for radiological and nonradiological traffic accidents. The probabilities, which are much lower than one, and the magnitudes of exposure were multiplied, yielding very low risk numbers. Incident-free risk factors were calculated for crew and public exposure to radiation emanating from the shipping container (cask) and public exposure to the chemical toxicity of the transportation vehicle exhaust. The probability of incident-free exposure is unity (one).

For each alternative, risks were assessed for both incident-free transportation and accident conditions. For the incident-free assessment, risks are calculated for both collective populations of potentially exposed individuals and for maximally exposed individuals. The accident assessment consists of two components: (1) a probabilistic accident risk assessment that considers the probabilities and consequences of a range of possible transportation accident environments, including low-probability accidents that have high consequences and high-probability accidents that have low consequences, and (2) an accident consequence assessment that considers only the consequences of the most severe postulated transportation accidents.

The RADTRAN 5 computer code (Neuhauser and Kanipe 1998) is used for incident-free and accident risk assessments to estimate the impacts on population. RADTRAN 5 was developed by Sandia National Laboratories to calculate population risks associated with the transportation of radioactive materials by a variety of modes, including truck, rail, air, ship, and barge. RADTRAN 5 was used to calculate the doses to the maximally exposed individuals.

The RADTRAN 5 population risk calculations include both the consequences and probabilities of potential exposure events. The RADTRAN 5 code consequence analyses include the cloud shine, ground shine, inhalation, and resuspension exposures. The collective population risk is a measure of the total radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing the various alternatives.

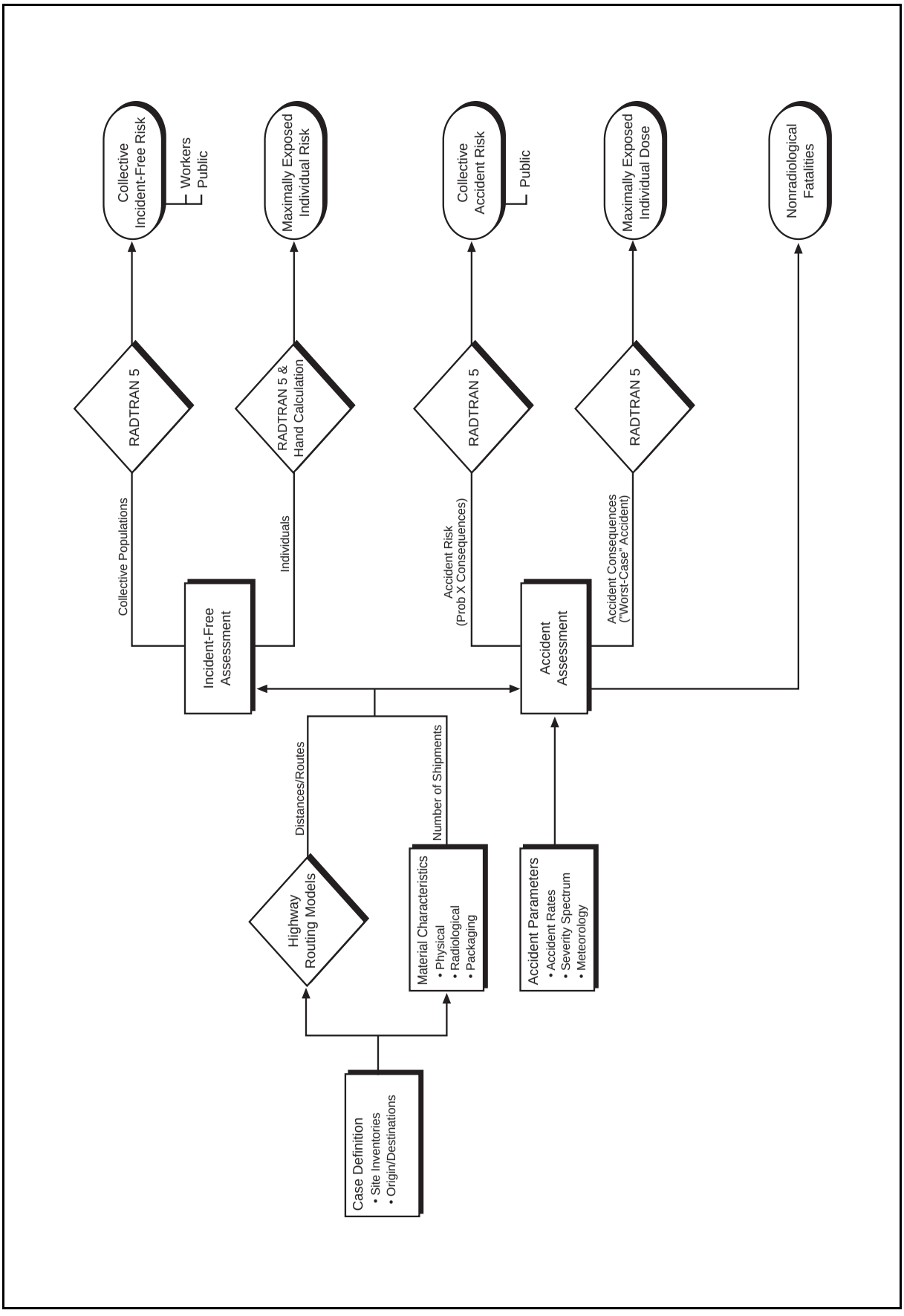


Figure G–4 Overland Transportation Risk Assessment

G.5 ALTERNATIVES, PARAMETERS, AND ASSUMPTIONS

G.5.1 Material Inventory and Shipping Campaigns

Table G–1 lists the fuels that could be shipped as a result of implementing an alternative to treat sodium-bonded spent nuclear fuel:

Table G–1 Transportation Summary for Sodium-Bonded Fuels

Fuel Type	*Applicable Alternatives*	*MTHM*	*Origin/ State*	*Destination/ State*	*Cask/Container Canister*	*Number of Shipments/ Type of Transport*
EBR-II Driver	All	1.1	ANL-W/ID	ANL-W/ID	HFEF-5	84/Onsite, intra-facility transfer
EBR-II Driver	1,2,3,4,5,6	2.0	INTEC/ID	ANL-W/ID	TN-FSV NAC-LWT	17 /Onsite With Roads Open 43/Onsite With Roads open
EBR-II Blanket	All	22.4	ANL-W/ID	ANL-W/ID	HFEF-5	165/Onsite, intra-facility transfer
Fast Flux Test Facility Driver	All	0.25	Hanford/ WA	ANL-W/ID	T-3	10/ Public Highways
Fermi-1 Blanket	All	34.2	INTEC/WA	ANL-W/ID	PB-1	14/Onsite with Road Closed
Miscellaneous	All	0.04	ORNL/TN SNL/NM	ANL-W/ID	Commercially Available Cask	1/Public Highways 1/Public Highways
Declad EBR-II Blanket	3,5	22.4	ANL-W/ID	SRS/SC	NAC-LWT	11/Public Highways
Declad Fermi-1 Blanket	3,5	34.2	ANL-W/ID	SRS/SC	NAC-LWT	18/Public Highways

MTHM = metric tons of heavy metal

The following shipment campaigns related to sodium-bonded spent nuclear fuel were analyzed by DOE in other NEPA documents and are not treated in detail here.

- Fast Flux Test Facility driver material is currently stored at the Hanford Site, and the transportation impacts are included in the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final EIS* (Programmatic Spent Nuclear Fuel EIS) (DOE 1995), and finalized in the Amendment to the Record of Decision (61 FR 9441).

- Miscellaneous spent nuclear fuel is currently stored at the Oak Ridge National Laboratory and at Sandia National Laboratory, and the transportation impacts are included in the Programmatic Spent Nuclear Fuel EIS (DOE 1995), and finalized in the Amendment to the Record of Decision (61 FR 9441).

The Programmatic Spent Nuclear Fuel EIS (DOE 1996) analyzed the transportation impacts in the same manner as this EIS. The representative routes were modeled using the HIGHWAY Code (Johnson, et. al. 1993), and the risks were quantified using an older version of the RADTRAN code. Since the Programmatic Spent Nuclear Fuel EIS assumed cask dose rates to be equal to the regulatory limit, and used representative fuel isotopics rather than actual sodium-bonded fuel isotopics, the published impact estimates were bounding.

All EBR-II blanket and some EBR-II driver 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 NRC regulations do not require the use of a certified Type B cask. DOE would use the HFEF-5 canister which is the sealed canister in which the spent nuclear fuel is currently stored. No incident-free risk analysis is necessary, because the public would receive no measurable exposure. Worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. No accident analysis is necessary because potential accidents during movement are bounded in frequency and consequence by handling accidents. Once the cask is closed for the low-speed movement to the nearby building, the likelihood and consequence of any foreseeable accident are very small and not further quantified.

Fermi-1 blanket fuel would be shipped from the INTEC to ANL-W in the formerly certified Type B cask, the PB-1 Cask. Since DOE would close the roads between INTEC and ANL-W using existing traffic gates, and there are no homes in the vicinity of the road within the INEEL site boundary, no quantitative analysis is necessary. No incident-free risk analysis is necessary, because the public would receive no measurable exposure. 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 the movement on the INEEL site roads, the likelihood and consequence of any foreseeable accident are very small.

EBR-II driver fuel currently stored at INTEC would be shipped to ANL-W in a certified Type B cask, either TN-FSV or NAC-LWT. Since the cask would be certified, DOE would not close the roads between INTEC and ANL-W. However, since there are no homes in the vicinity of the road within the INEEL site boundary, limited quantitative analysis is necessary. No incident-free risk analysis for exposure to the public at stops or in their homes is necessary. 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 the movement on the INEEL site roads, the likelihood and consequence of any foreseeable accident are very small and not further quantified.

G.5.2 Representative Routes

Representative overland truck routes were selected for the shipments from ANL-W to SRS. The routes were selected consistent with current routing practices and all applicable routing regulations and guidelines (DOT 1992). However, the routes were determined for risk assessment purposes. They do not necessarily represent the actual routes that would be used to transport spent nuclear fuel in the future. Specific routes cannot be identified in advance. The representative truck routes are shown in **Figure G–5**.

Route characteristics that are important to the radiological risk assessment include the total shipment distance and the population distribution along the route. The specific route selected determines both the total potentially exposed population and the expected frequency of transportation-related accidents. Route characteristics are summarized in **Table G–2**. The population densities along each route are derived from 1990 U.S. Bureau of Census data. Rural, suburban, and urban areas are characterized according to the following breakdown: rural population densities range from 0 to 54 persons per square kilometer (0 to 139 person per square mile); the suburban range is from 55 to 1,284 persons per square kilometer (140 to 3,326 persons per square mile); and the urban range includes all population densities greater than 1,284 persons per square kilometer (3,326 persons per square mile). The exposed population includes all persons living within 800 meters (0.5 mile) of each side of the road. The exposed population, for route characterization and incident-free dose calculation, includes all persons living within 800 meters (0.5 mile) of each side of the road.

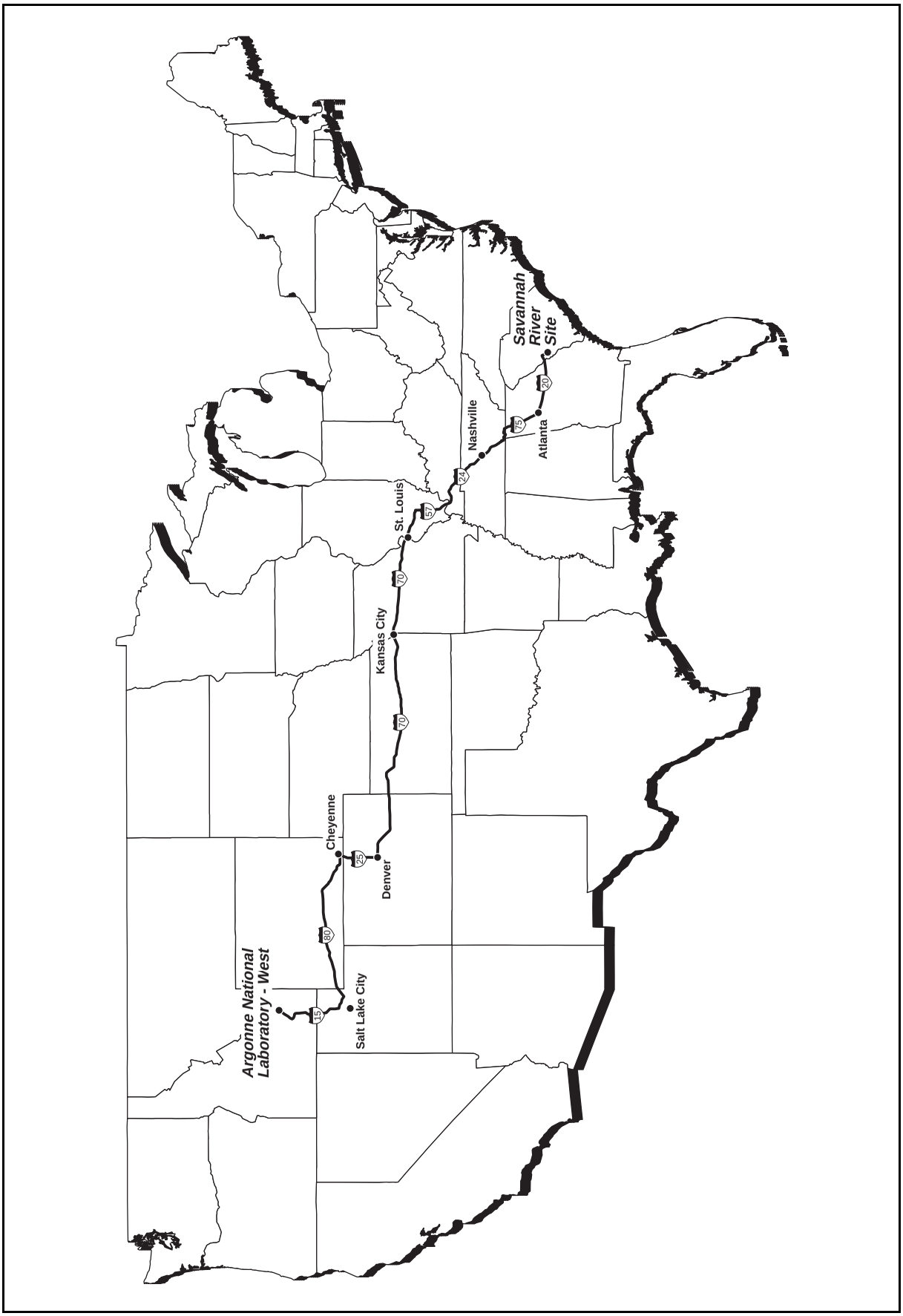


Figure G–5 Representative Overland Truck Route

Table G–2 Potential Shipping Routes Evaluated for the Sodium Bonded Spent Nuclear Fuel EIS

From	*To*	*Distance (km)*	*Percentages in Zones*			*Population Density in Zone ($1/km^2$)*			*Number of Affected Persons*
			Rural	*Suburban*	*Urban*	*Rural*	*Suburban*	*Urban*	
Truck Routes									
ANL-W	SRS	3759.3	82.8	15.4	1.8	7.4	353	2173.3	599,000
INTEC	ANL-W	38.6	100	0	0	1.0	N/A	N/A	62

km = kilometers, N/A = not applicable
The shipment impact to SRS are all based on the distance and population exposed on a trip from ANL-W to SRS.

G.5.3 External Dose Rates

External dose rates are calculated for the spent nuclear fuel being shipped on public roads (SAIC 1999). For the EBR-II blanket fuel, the dose rate on contact with the cask is 0.6 millirem per hour and the dose rate at 2 meters (6 feet) from the cask is 0.1 millirem per hour. For the Fermi-1 blanket fuel, the dose rate on contact with the cask is 7.1×10^{-4} millirem per hour and the dose rate at 2 meters (6 feet) from the cask is 1.4×10^{-4} millirem per hour. For the EBR-II driver material shipped to ANL-W, the dose rate on contact with the cask is 0.59 millirem per hour and the dose rate at 2 meters (6 feet) from the cask is 0.12 millirem per hour.

G.5.4 Health Risk Conversion Factors

The health risk conversion factors used to estimate expected cancer fatalities were: 0.0005 and 0.0004 fatal cancer cases per person-rem for members of the public and workers, respectively (NCRP 1993).

G.5.5 Accident Frequencies

For the calculation of accident risks, vehicle accident and fatality rates are taken from data provided in other reports (ANL 1994). Accident rates are generically defined as the number of accident involvements (or fatalities) in a given year per unit of travel in that same year. Therefore, the rate is a fractional value, with accident-involvement count as the numerator of the fraction and vehicular activity (total travel distance in truck-kilometers) as its denominator. Accident rates are generally determined for a multi-year period. For assessment purposes, the total number of expected accidents or fatalities is calculated by multiplying the total shipment distance for a specific case by the appropriate accident or fatality rate.

For truck transportation, the rates presented are specifically for heavy combination trucks involved in interstate commerce (ANL 1994). Heavy combination trucks are rigs composed of a separable tractor unit containing the engine and one to three freight trailers connected to each other. Heavy combination trucks are typically used for radioactive waste shipments. The truck accident rates are computed for each state based on statistics from 1986 to 1988 compiled by the U.S. Department of Transportation Office of Motor Carriers. Saricks and Kvitek (ANL 1994) present accident involvement and fatality counts; estimated kilometers of travel by state; and the corresponding average accident involvement, fatality, and injury rates for the three years investigated. A fatality caused by an accident is the death of a member of the public who is killed instantly or dies within 30 days due to the injuries sustained in the accident.

G.5.6 Container Accident Response Characteristics and Release Fractions

G.5.6.1 Development of Conditional Probabilities

NUREG-0170 (NRC 1977) originally was used to estimate the conditional probabilities associated with the accidents involving transportation of radioactive materials. The Modal Study, an initiative taken by the NRC (NRC 1987) to refine more precisely the analysis presented in NUREG-0170 (NRC 1977) for spent nuclear fuel shipping casks, was used to estimate the conditional probabilities of accidents.

Whereas the NUREG-0170 analysis was primarily performed using best engineering judgments and presumptions concerning cask response, the Modal Study relies on sophisticated structural and thermal engineering analysis and a probabilistic assessment of the conditions that could be experienced in severe transportation accidents. The Modal Study results are based on representative spent nuclear fuel casks assumed to have been designed, manufactured, operated, and maintained according to national codes and standards. Design parameters of the representative casks were chosen to meet the minimum test criteria specified in 10 CFR 71. The study is believed to provide realistic, yet conservative, results for radiological releases under transport accident conditions.

In the Modal Study, potential accident damage to a cask is categorized according to the magnitude of the mechanical forces (impact) and thermal forces (fire) to which a cask may be subjected during an accident. Because all accidents can be described in these terms, severity is independent of the specific accident sequence. In other words, any sequence of events that results in an accident in which a cask is subjected to forces within a certain range of values is assigned to the accident severity region associated with that range. The accident severity scheme is designed to take into account all potential foreseeable transportation accidents, including accidents with low probability but high consequences, and those with high probability but low consequences.

As discussed above, the accident consequence assessment only considers the potential impacts from the most severe transportation accidents. In terms of risk, the severity of an accident must be viewed in terms of potential radiological consequences, which are directly proportional to the fraction of the radioactive material within a cask that is released to the environment during the accident. Although regions span the entire range of mechanical and thermal accident loads, they are grouped into accident categories that can be characterized by a single set of release fractions and are, therefore, considered together in the accident consequence assessment. The accident category severity fraction is the sum of all conditional probabilities in that accident category.

G.5.6.2 Release Fraction Assumptions

The release fractions for were taken from the Programmatic Spent Nuclear Fuel EIS (DOE 1995), which was based on the above described Modal Study. Spent nuclear fuel could be shipped in two different forms: unaltered or declad. The construction and cladding of the spent nuclear fuel are assumed to be similar enough to aluminum-clad fuels analyzed in that EIS that the performance in an accident would be similar. The declad fuel would also exhibit similar performance, since the fuel is placed in a shipping can which is in turn placed inside the transportation cask.

G.5.7 Nonradiological Risk (Vehicle Related)

Vehicle-related health risks resulting from incident-free transport may be associated with the generation of air pollutants by transport vehicles during shipment and are independent of the radioactive nature of the shipment. The health end-point assessed under incident-free transport conditions is the excess latent mortality due to inhalation of vehicle exhaust emissions. Risk factors for pollutant inhalation in terms of latent mortality have been generated (Neuhauser and Kanipe 1998). These risks are 1×10^{-7} mortality per kilometer (1.6×10^{-7} per

mile) of truck travel in urban areas. The risk factors are based on regression analyses of the effects of sulfur dioxide and particulate releases from diesel exhaust on mortality rates. Excess latent mortalities are assumed to be equivalent to latent cancer fatalities. Vehicle-related risks from incident-free transportation are calculated for each case by multiplying the total distance traveled in urban areas by the appropriate risk factor. Similar data are not available for rural and suburban areas.

Risks are summed over the entire route and over all shipments for each case. This method has been used in several EISs to calculate risks from incident-free transport. Lack of information for rural and suburban areas is an obvious data gap, although the risk factor would presumably be lower than for urban areas because of lower total emissions from all sources and lower population densities in rural and suburban areas.

G.6 RISK ANALYSIS RESULTS

Per-shipment risk factors have been calculated for the collective populations of exposed persons and for the crew for all anticipated routes and shipment configurations. The radiological risks are presented in doses per shipment for each unique route, material, and container combination. The radiological dose per shipment factors for incident-free transportation are presented in **Table G–3** for the transportation routes analyzed for this EIS. As stated in Section G.5.1, the Programmatic Spent Nuclear Fuel EIS (DOE 1996) used very conservative assumptions to analyze the shipments from the Oak Ridge Reservation, Hanford Site, and Sandia National Laboratory. For these 12 shipments, the incident free public risk is 9.7×10^{-4} latent cancer fatalities from radiation and 8.1×10^{-6} latent cancer fatalities from exhaust emissions. The crew radiological risk is 3.1×10^{-4} latent cancer fatalities. The public risk from radiological accidents is 4.0×10 latent cancer fatalities and from nonradiological accidents is 1.2×10^{-3} fatalities.

Doses are calculated for the crew, off-link public (i.e., people living along the route), on-link public (i.e., pedestrians and drivers along the route), and public at rest and fueling stops (i.e., stopped cars, buses and trucks, workers, and other bystanders). For the onsite shipments from INTEC to ANL-W, the stop dose is set to zero, because a truck would not be expected to stop during a trip that takes less than an hour. The off-link dose is zero because no persons are residing within 800 meters (0.5 miles) of the road.

The radiological dose risk factors for transportation accidents are also presented in Table G–3. The accident risk factors are called "dose risk" because the values incorporate the spectrum of accident severity probabilities and associated consequences. The accident dose is very low because, although persons are residing in an 80 kilometers (50 miles) radius of the road, they are generally quite far from the road. Since RADTRAN 5 uses an assumption of homogeneous population from the road out to 80 kilometers (50 miles), it greatly overestimates the actual doses. However, the doses are clearly several factors of ten lower than the doses for the other transportation legs shown in Table G–3.

The nonradiological risk factors are presented in fatalities per shipment in **Table G–4**. Separate risk factors are provided for fatalities resulting from exhaust emissions (caused by hydrocarbon emissions known to be carcinogens) and transportation accidents (fatalities resulting from impact).

Table G–5 shows the risks of transportation for each alternative. The risks are calculated by multiplying the previously given per-shipment factors by the number of shipments over the duration of the program and, for the radiological doses, by the health risk conversion factors. The incident-free doses from the onsite shipments are very high, relative to the distance traveled, because the regulatory limit dose rate was used. As previously stated, the calculated dose rates for the packages being shipped to SRS are several factors of 10 lower than the regulatory limit.

Table G–3 Radiological Risk Factors for Single Shipments

From	To	Material & Package	Incident-Free Dose (Person-rem)					Accident Dose (Person-rem)
			Crew	Public				
				Off-link	On-link	Stops	Total	
ANL	SRS	EBR-II Blanket	1.07E-04	1.74E-04	9.02E-04	3.25E-07	1.08E-03	2.71E-7
ANL	SRS	Fermi-1 Blanket	1.34E-07	2.18E-07	1.13E-06	4.06E-10	1.35E-06	3.55E-9
INTEC	ANL-W	EBR-II Driver	1.10E-06	0.00E+00	8.10E-06	0.00E+00	8.10E-06	less than 1E-10

Table G–4 Nonradiological Risk Factors per Shipment

Nonradiological Risk Estimates (Fatalities/Shipment)			
From	To	Exhaust Emission	Accident
ANL-W	SRS	2.8E-5	7.0E-4
INTEC	ANL-W	0	8.2E-6

Table G–5 Risks of Transporting the Hazardous Materials[a]

Material Shipped	Alternative	Distance on Public Roads (kilometers)	Incident-Free				Accident
			Radiological (person-rem)		Nonradiological		
			Crew	Public	Emission	Traffic	Radiological
None[b]	No Action	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
EBR-II Driver Fuel	1	1,660	1.89E-08	1.74E-07	0.00E+00	8.21E-07	less than 1E-12
EBR-II Driver Fuel	2	1,660	1.89E-08	1.74E-07	0.00E+00	8.21E-07	less than 1E-12
EBR-II and Declad and Cleaned EBR-II and Fermi-1 Blanket Fuel	3	110,680	7.86E-07	6.09E-06	1.96E-04	2.11E-03	1.62E-09
EBR-II Driver Fuel	4	1,660	1.89E-08	1.74E-07	0.00E+00	8.21E-07	less than 1E-12
EBR-II Driver and Declad and Cleaned EBR-II and Fermi-1 Blanket Fuel	5	110,680	7.86E-07	6.09E-06	1.96E-04	2.11E-03	1.62E-09
EBR-II Driver Fuel	6	1,660	1.89E-08	1.74E-07	0.00E+00	8.21E-07	less than 1E-12

[a] All risks are expressed as number of latent cancer fatalities, except for the Accident-Traffic column, which lists number of accident fatalities.

[b] It is assumed that no material analyzed in this EIS would be shipped. If DOE would choose to move the EBR-II Driver material currently in storage at INTEC to ANL-W, the transportation impacts would be the same as those calculated for Alternatives 1, 2, 3, and 5.

The risks to various exposed individuals under incident-free transportation conditions have been estimated for hypothetical exposure scenarios. The estimated doses to workers and the public are presented in **Table G–6.**

Table G–6 Estimated Dose to Exposed Individuals During Incident-Free Transportation Conditions

Receptor		*Dose to Maximally Exposed Individual*	
		Idaho to SRS	***Intrasite***
Workers	Crew member (truck driver) [a]	1.5×10^{-6} rem	1.2×10^{-6} rem
	Inspector	2.9×10^{-5} rem per event	Not Applicable
Public	Resident	4.0×10^{-9} rem per event	Not Applicable
	Person in traffic congestion	1.1×10^{-4} rem per event	0.003 rem per event
	Person at service station	1.0×10^{-5} rem per event	Not Applicable

[a] Assumes that an individual driver takes every shipment.

All doses are presented on a per-event basis (person-rem per event) because it is not likely that the same person will be exposed to multiple events. The dose to the maximally exposed crew member is based on the same individual being responsible for driving every shipment for the duration of the campaign. Note that the potential exists for larger individual exposures if multiple exposure events occur. For example, the dose to a person stuck in traffic next to a shipment for 10 minutes is calculated to be 3 millirem. However, since the intersite shipments pass through urban areas, a 30-minute exposure time is considered. Using the estimated dose rates, the maximally exposed individual would receive 0.1 millirem. If the exposure duration were longer, the dose would rise proportionally. In addition, a person working at a truck service station could receive a significant dose if trucks were to use the same stops repeatedly. The dose to a person fueling a truck could be as much as 0.01 millirem per event.

The cumulative dose to a resident was calculated assuming all shipments passed his or her home. The cumulative doses assume that the resident is present for every shipment and is unshielded at a distance of 30 meters (about 66 feet) from the route. Therefore, the cumulative dose depends on the number of shipments passing a particular point and is independent of the actual route being considered. The maximum dose to this resident, if all the material were to be shipped via this route, would be less than 0.01 millirem.

The estimated dose to transportation crew members is presented for a commercial crew. No credit is taken for the shielding associated with the tractor or trailer.

The accident consequence assessment is intended to provide an estimate of the maximum potential impacts posed by the most severe potential transportation accidents involving a shipment. The maximum foreseeable (frequency greater than 1×10^{-7} per year) offsite transportation accident involves a shipment of EBR-II blanket fuel material under neutral (average) weather conditions. The accident has a probability of occurrence of about 1 every 10 million years and could result in 0.46 person-rem to the public. Additionally the accident could result in a dose of 1.9×10^{-3} rem to the hypothetical maximally exposed individual in the immediate vicinity of the accident. The probability of an accident occurring and the exposed populations are lower for the onsite shipment of EBR-II blanket fuel. The source term is lower for the offsite shipments of Fermi blanket fuel. This accident would fall into Category 5 of the Modal Study accident matrix (NRC 1987), and would occur in a suburban population zone. To incur this level of damage, the cask would have to collide with an immovable object at a speed of much greater than 88 kilometers per hour (55 miles per hour). The probability of an accident with a more energetic collision or a significant fire, which could lead to higher consequences, is lower.

G.7 CONCLUSIONS AND LONG-TERM IMPACTS OF TRANSPORTATION

G.7.1 Conclusions

It is unlikely that the transportation of radioactive materials will cause an additional fatality.

G.7.2 Long-Term Impacts of Transportation

The Programmatic Spent Nuclear Fuel EIS (DOE 1995) analyzed the cumulative impacts of all transportation of radioactive materials, including impacts from reasonably foreseeable 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 summarized in **Table G–7**. The table shows that 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). Total general population collective doses were also estimated to be 320,000 person-rem (160 latent cancer fatalities). The majority of the collective dose for workers and the general population was due to the general transportation of radioactive material. Examples of these activities are shipments of radiopharmaceuticals to nuclear medicine laboratories and shipments of commercial low-level radioactive waste to commercial disposal facilities. The total number of latent cancer fatalities estimated to result from radioactive materials transportation over the period between 1943 and 2035 was 290. Over this same period (93 years), approximately 28 million people would die from cancer, based on 300,000 cancer fatalities per year. It should be noted that the estimated number of transportation-related latent cancer fatalities would be indistinguishable from other latent cancer fatalities, and the transportation-related latent cancer fatalities are 0.0010 percent of the total number of latent cancer fatalities.

Table G–7 Cumulative Transportation-Related Radiological Collective Doses and Latent Cancer Fatalities (1943 to 2035)

Category	*Collective Worker Dose (person-rem)*	*Collective General Population Dose (person-rem)*
Sodium-bonded Fuel Impacts (from Table G–4)	less than 1	less than 1
Other Nuclear Material Shipments		
Truck	11,000	50,000
Rail	820	1,700
General transportation (1943–2035)	310,000	270,000
Total collective dose	320,000	320,000
Total latent cancer fatalities	130	160

Source: DOE 1995.

G.8 UNCERTAINTY AND CONSERVATISM IN ESTIMATED IMPACTS

The sequence of analyses performed to generate the estimates of radiological risk for transportation includes: (1) determination of the inventory and characteristics, (2) estimation of shipment requirements, (3) determination of route characteristics, (4) calculation of radiation doses to exposed individuals (including estimating of environmental transport and uptake of radionuclides), and (5) estimation of health effects. Uncertainties are associated with each of these steps. Uncertainties exist in the way that the physical systems being analyzed are represented by the computational models; in the data required to exercise the models (due to measurement errors, sampling errors, natural variability, or unknowns simply caused by the future nature of the actions being analyzed); and in the calculations themselves (e.g., approximate algorithms used by the computers).

In principle, one can estimate the uncertainty associated with each input or computational source and predict the resultant uncertainty in each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final, or absolute, result; however, conducting such a full-scale quantitative uncertainty analysis is often impractical and sometimes impossible, especially for actions to be initiated at an unspecified time in the future. Instead, the risk analysis is designed to ensure, through uniform and judicious selection of scenarios, models, and input parameters, that relative comparisons of risk among the various alternatives are meaningful. In the transportation risk assessment, this design is accomplished by uniformly applying common input parameters and assumptions to each alternative. Therefore, although considerable uncertainty is inherent in the absolute magnitude of the transportation risk for each alternative, much less uncertainty is associated with the relative differences among the alternatives in a given measure of risk.

In the following sections, areas of uncertainty are discussed for the assessment steps enumerated above. Special emphasis is placed on identifying whether the uncertainties affect relative or absolute measures of risk. The reality and conservatism of the assumption are addressed. Where practical, the parameters that most significantly affect the risk assessment results are identified.

G.8.1 Uncertainties in Material Inventory and Characterization

The inventories and the physical and radiological characteristics are important input parameters to the transportation risk assessment. The potential amount of transportation for any alternative is determined primarily by the projected dimensions of package contents, the strength of the radiation field, the heat that must be dissipated, and assumptions concerning shipment capacities. The physical and radiological characteristics are important in determining the material released during accidents and the subsequent doses to exposed individuals through multiple environmental exposure pathways.

Uncertainties in the inventory and characterization will be reflected in the transportation risk results. If the inventory is overestimated (or underestimated), the resulting transportation risk estimates also will be overestimated (or underestimated) by roughly the same factor. However, the same inventory estimates are used to analyze the transportation impacts of each of the EIS alternatives. Therefore, for comparative purposes, the observed differences in transportation risks among the alternatives, as given in Table G-5, are believed to represent unbiased, reasonably accurate estimates from current information in terms of relative risk comparisons.

G.8.2 Uncertainties in Containers, Shipment Capacities, and Number of Shipments

The transportation required for each alternative is based in part on assumptions concerning the packaging characteristics and shipment capacities for commercial trucks and safe secure transports. Representative shipment capacities have been defined for assessment purposes based on probable future shipment capacities. In reality, the actual shipment capacities may differ from the predicted capacities such that the projected number of shipments and, consequently, the total transportation risk would change. However, although the predicted transportation risks would increase or decrease accordingly, the relative differences in risks among alternatives would remain about the same.

G.8.3 Uncertainties in Route Determination

Representative routes have been determined between all origin and destination sites considered in the EIS. The routes have been determined to be consistent with current guidelines, regulations, and practices, but may not be the actual routes that would be used in the future. In reality, the actual routes could differ from the representative ones concerning distances and total population along the routes. Moreover, since materials could be transported over an extended time starting at some time in the future, the highway infrastructures and

the demographics along routes could change. These effects have not been accounted for in the transportation assessment; however, it is not anticipated that these changes would significantly affect relative comparisons of risk among the alternatives considered in the EIS. Specific routes cannot be identified in advance because the routes are classified to protect national security interests.

G.8.4 Uncertainties in the Calculation of Radiation Doses

The models used to calculate radiation doses from transportation activities introduce a further uncertainty in the risk assessment process. Estimating the accuracy or absolute uncertainty of the risk assessment results is generally difficult. The accuracy of the calculated results is closely related to the limitations of the computational models and to the uncertainties in each of the input parameters that the model requires. The single greatest limitation facing users of RADTRAN, or any computer code of this type, is the scarcity of data for certain input parameters.

Uncertainties associated with the computational models are reduced by using state-of-the-art computer codes that have undergone extensive review. Because many uncertainties are recognized but difficult to quantify, assumptions are made at each step of the risk assessment process intended to produce conservative results (i.e., overestimate the calculated dose and radiological risk). Because parameters and assumptions are applied to all alternatives, this model bias is not expected to affect the meaningfulness of relative comparisons of risk; however, the results may not represent risks in an absolute sense.

Post accident mitigative actions are not considered for dispersal accidents. For severe accidents involving the release and dispersal of radioactive materials in the environment, no post accident mitigative actions, such as interdiction of crops or evacuation of the accident vicinity, have been considered in this risk assessment. In reality, mitigative actions would take place following an accident according to U.S. Environmental Protection Agency radiation protection guides for nuclear incidents (EPA 1991). The effects of mitigative actions on population accident doses are highly dependent upon the severity, location, and timing of the accident. For this risk assessment, ingestion doses are only calculated for accidents occurring in rural areas (the calculated ingestion doses, however, assume all food grown on contaminated ground is consumed and is not limited to the rural population). Examination of the severe accident consequence assessment results has shown that ingestion of contaminated foodstuffs contributes about 50 percent of the total population dose for rural accidents. Interdiction of foodstuffs would act to reduce, but not eliminate, this contribution.

G.9 REFERENCES

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APPENDIX H
ENVIRONMENTAL JUSTICE ANALYSIS

H.1 INTRODUCTION

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, directs Federal agencies to identify and address, as appropriate, the disproportionately high and adverse health or environmental effects of their programs, policies, and activities on minority populations and low-income populations.

The Council on Environmental Quality has oversight responsibility for documentation prepared in compliance with the National Environmental Policy Act (NEPA). In December 1997, the Council released its guidance on environmental justice under NEPA (CEQ 1997). The Council's guidance was adopted as the basis for the analysis of environmental justice contained in this environmental impact statement (EIS).

This appendix provides an assessment of the potential for disproportionately high and adverse human health or environmental effects on minority or low-income populations that could result from implementation of alternatives for management of the U.S. Department of Energy's (DOE) inventory of sodium-bonded spent nuclear fuel.

H.2 DEFINITIONS AND APPROACH

Minority Individuals and Population

The following definitions of minority individuals and population were used in this analysis of environmental justice:

- **Minority Individuals**—Members of any of the following population groups: Hispanic, Native American, Asian or Pacific Islander, or Black

- **Minority Population**—The total number of minority individuals residing within a potentially affected area

In discussions of environmental justice in this EIS, persons self-designated as Hispanic are included in the Hispanic population, regardless of race. For example, the Asian or Pacific Islander population is composed of persons self-designated as Asian or Pacific Islander and not of Hispanic origin. Asian or Pacific Islanders who designate themselves as having Hispanic origins are included in the Hispanic population. Data for the analysis of minorities and racial population were extracted for the year 2010 from the U.S. Census Bureau's worldwide web site (DOC 1999).

Executive Order 12898 specifically addresses "disproportionately high and adverse effects" on "low-income" populations. The Council on Environmental Quality recommends that poverty thresholds be used to identify "low-income" individuals (CEQ 1997).

Low-Income Individuals and Population

The following definitions of low-income individuals and population were used in this analysis:

- **Low-Income Individuals**—Persons whose self-reported incomes are less than the poverty threshold
- **Low-Income Population**—The total number of poverty-level individuals residing within a potentially affected area

Data for the analysis of low-income populations were extracted from the U.S. Census Bureau's Table P121 of Standard Tape File 3 (DOC 1992).

Disproportionately High and Adverse Human Health Effects

Adverse health effects are measured in risks and rates that could result in latent cancer fatalities, as well as other fatal or nonfatal adverse impacts to human health. Disproportionately high and adverse human health effects occur when the risk or rate of exposure to an environmental hazard for a minority or low-income population is significant and exceeds the risk of exposure rate for the general population or, where available, for another appropriate comparison group (CEQ 1997).

Disproportionately High and Adverse Environmental Impacts

A disproportionately high environmental impact refers to an impact or risk of an impact in a low-income or minority community that is significant and exceeds the environmental impact on the larger community. An adverse environmental impact is a deleterious environmental impact that is determined to be significant. In assessing cultural and aesthetic environmental impacts, impacts that uniquely affect geographically dislocated or dispersed low-income or minority populations were considered (CEQ 1997).

Potentially affected areas examined in this EIS include areas defined by an 80-kilometer (50-mile) radius centered on candidate facilities for the treatment and management of sodium-bonded spent nuclear fuel at Argonne National Laboratory-West (ANL-W) and the Savannah River Site (SRS).

H.3 METHODOLOGY

H.3.1 Spatial Resolution

For the purposes of enumeration and analysis, the U.S. Census Bureau has defined a variety of areal units (DOC 1992). Areal units of concern in this EIS include (in order of increasing spatial resolution) states, counties, census tracts, block groups, and blocks. The block is generally the smallest of these entities and offers the finest spatial resolution. This term refers to a relatively small geographical area bounded on all sides by visible features such as streets and streams or by invisible boundaries such as city limits and property lines. During the 1990 census, the U.S. Census Bureau subdivided the United States and its territories into 7,017,425 A blocks. For comparison, the number of counties, census tracts, and block groups used in the 1990 census were 3,248; 62,276; and 229,192, respectively. While blocks offer the finest spatial resolution, economic data required for identification of low-income populations are not available at the block level of spatial resolution. In the analysis below, block groups are used throughout as the areal unit. Block groups generally contain between 250 and 500 housing units (DOC 1992).

During the decennial census, the U.S. Census Bureau collects data from individuals and aggregates the data according to residence in a geographical area, such as a county or block group. Boundaries of the areal units are selected to coincide with features such as streams and roads or political boundaries such as county and city

borders. Boundaries used for aggregation of the census data usually do not coincide with boundaries used in the calculation of health effects. As discussed in Chapter 4 of this EIS, radiological health effects due to an accident at each of the sites are evaluated for persons residing within a distance of 80 kilometers (50 miles) of the accident site. In general, the boundary of the circle with an 80-kilometer (50-mile) radius centered at the accident site will not coincide with boundaries used by the U.S. Census Bureau for enumeration of the population in the potentially affected area. Some block groups lie completely inside or outside of the radius for health effects calculation. However, other block groups are only partially included. As a result of these partial inclusions, uncertainties are introduced into the estimate of the population at risk from the accident.

To estimate the populations at risk in partially included block groups, it was assumed that populations are uniformly distributed throughout the area of each block group. For example, if 30 percent of the area of a block group lies within 80 kilometers (50 miles) of the accident site, it was assumed that 30 percent of the population residing in that block group would be at risk. An upper bound for the population at risk was obtained by including the total population of partially included block groups in the population at risk. Similarly, a lower bound for the population at risk was obtained by excluding the population of partially included blocks from the population at risk. As a general rule, if the areas of geographic units defined by the U.S. Census Bureau are small in comparison with the potentially affected area, then the uncertainties due to partial inclusions will be relatively small.

H.3.2 Population Projections

Health effects were calculated for populations projected to reside in potentially affected areas during the year 2010. Extrapolations of the total population for individual states are available from both the U.S. Census Bureau and various state agencies (Campbell 1996). The U.S. Census Bureau also projects populations by ethnic and racial classification in one-year intervals for the years from 1995 to 2025 at the state level. State agencies project total populations for individual counties. No Federal or state agency projects block groups or low-income populations. Data used to project minority populations were extracted from the U.S. Census Bureau's Internet web site (DOC 1999). To project minority populations in potentially affected areas, minority populations determined from the 1990 census data were taken as a baseline for each block group. Then it was assumed that percentage changes in the minority population of each block group for a given year (compared to the 1990 baseline data) will be the same as percentage changes in the state minority population projected for the same year. An advantage to this assumption is that the projected populations are obtained using a consistent method, regardless of the state and associated block group involved in the calculation. A disadvantage is that the method is insensitive to localized demographic changes that could alter the projection in a specific area.

The U.S. Census Bureau uses the cohort-component method to estimate future populations for each state (Campbell 1996). The set of cohorts is composed of: (1) age groups from 1 year or less to 85 years or more, (2) male and female populations in each age group, and (3) the following racial and ethnic groups in each age group: Hispanic, non-Hispanic Asian, non-Hispanic African American, non-Hispanic Native American, and non-Hispanic White. Components of the population change used in the demographic accounting system are births, deaths, net state-to-state migration, and net international migration. If P(t) denotes the number of individuals in a given cohort at time "t," then:

$$P(t) = P(t_0) + B - D + DIM - DOM + IIM - IOM \quad (1)$$

where:

$P(t_0)$ = Cohort population at time t_0 is less than or equal to t. For this analysis, t_0 denotes the year 1990.

B = Births expected during the period from t_0 to t.

D = Deaths expected during the period from t_0 to t.
DIM = Domestic migration into the state expected during the period from t_0 to t.
DOM = Domestic migration out of the state expected during the period from t_0 to t.
IIM = International migration into the state expected during the period from t_0 to t.
IOM = International migration out of the state expected during the period from t_0 to t.

Estimated values for the components shown on the right side of the equation are based on past data and various assumptions regarding changes in the rates for birth, mortality, and migration (Campbell 1996). Persons of Hispanic origin are included in the Hispanic population regardless of race. It should be noted that the U.S. Census Bureau does not project populations of individuals who identified themselves as "other race" during the 1990 census. This population group is less than 2 percent of the total population in each of the states. However, to project total populations in the environmental justice analysis, population projections for the "other race" group were made under the assumption that the growth rate for the "other race" population will be identical to the growth rate for the combined minority and White populations.

H.4 ENVIRONMENTAL JUSTICE ASSESSMENT

The analysis of environmental justice effects was based on an assessment of the impacts reported in Chapter 4 of this EIS. This analysis was performed to identify any disproportionately high and adverse human health or environmental impacts on minority or low-income populations surrounding ANL-W and SRS. Demographic information obtained from the U.S. Census Bureau was used to identify the minority populations and low-income communities in the zone of potential impact surrounding the two sites. The zone, or region of influence, is a circle that has an 80-kilometer (50-mile) radius around the proposed sites. This radius is consistent with that used to evaluate the collective dose for human health effects, air impact modeling, and socioeconomic impacts, and is judged to encompass all of the impacts that may occur.

H.5 RESULTS FOR THE SITES

As discussed in Chapter 2 of this EIS, candidate sites for the treatment and management of sodium-bonded spent nuclear fuel are located at ANL-W and SRS. This section describes the environmental justice analysis of potentially affected minority and low-income populations residing near the candidate sites. It should be noted that projections of the total population provided in this appendix differ from the projected total populations used in the health effects calculations described in Chapter 4. This is because the projections used in the analysis of environmental justice are based on projections for the states provided by the U.S. Bureau of the Census (Campbell 1996). Projections used in the analysis of health effects are based on county-wide projections provided by state agencies. As discussed in Section H.3.2, the county projections are more sensitive to localized demographic changes. However, the states do not provide projections for minority populations. Therefore, the U.S. Bureau of the Census projections were used in the analysis of environmental justice. Population projections obtained with the two approaches differ by 8 percent or less and have essentially no effect on these results of the analyses.

H.5.1 Argonne National Laboratory-West

Figure H–1 shows the racial and ethnic composition of the minority population of ANL-W projected to reside in the potentially affected area in the year 2010. In the interval between 1990 and 2010, the percentage of the total population composed of minorities is projected to increase from 8.7 percent to 13.3 percent. For comparison, during the 1990 census, minorities were found to compose approximately one-quarter of the total national population. By the year 2010, minorities are projected to compose closer to one-third of the total national population. The percentage of the minority population residing in the potentially affected area surrounding ANL-W was less than the corresponding national percentage in 1990, and is expected to remain so through the year 2010. Hispanics are the largest minority group residing in the potentially affected area, while the Asian and Hispanic populations are projected to show the largest growth rates.
Figure H–2 shows the location of minority populations residing near the ANL-W in 1990. As indicated in the

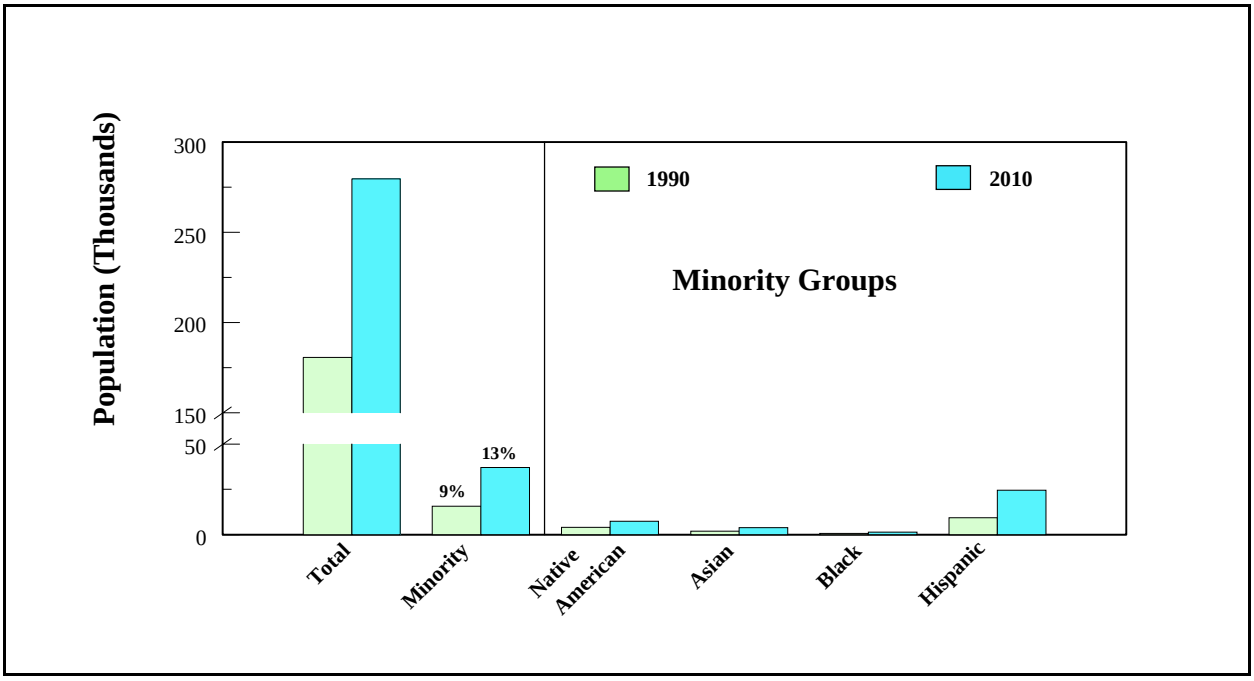

Figure H–1 Projected Racial and Ethnic Composition of the Minority Population Residing Within 80 Kilometers (50 Miles) of ANL-W in 2010

figure, block groups for which the percentage of minority residents exceeds the corresponding national percentage are located throughout the potentially affected area.

During the 1990 census, 15 percent of the residents within the potentially affected area surrounding ANL-W reported incomes below the poverty threshold. Slightly over 13 percent of the national population reported incomes below the poverty threshold, and approximately 13 percent of the residents of Idaho reported incomes below the poverty threshold during the same year. Thus, the percentage of the low-income population residing within the potentially affected area exceeded that for the nation and the state of Idaho by approximately 2 percent. **Figure H–3** shows the geographical distribution of low-income residents surrounding the ANL-W site. Block groups for which the percentage of low-income residents exceeds the corresponding national percentage are located throughout the potentially affected area.

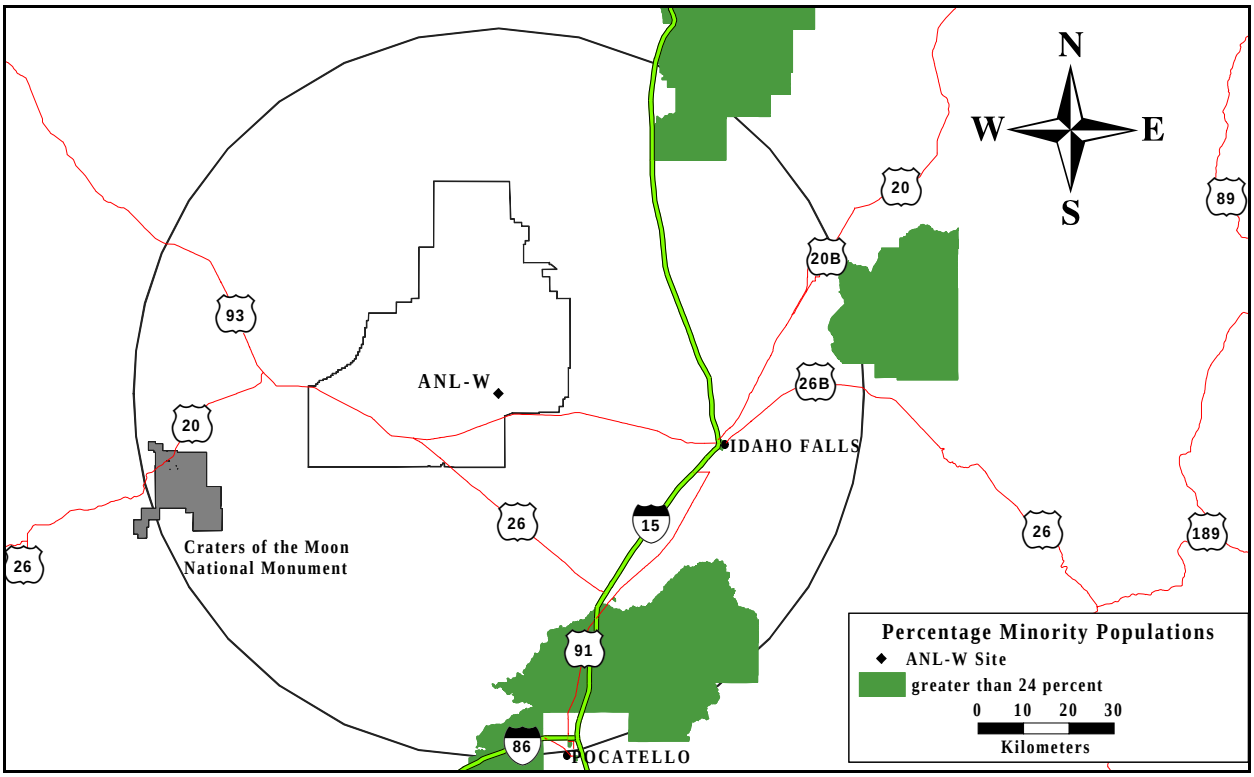

Figure H-2 Minority Population Residing Within 80 Kilometers (50 Miles) of the ANL-W Site in 1990

As discussed in Chapter 4, implementation of the alternatives at ANL-W would pose little risk to the public and the natural environment. Thus, selection of the alternatives that result in activities at ANL-W for the treatment and management of sodium-bonded spent nuclear fuel would not be expected to pose disproportionately high and adverse risks to potentially affected minority and low-income populations residing near ANL-W.

H.5.2 The Savannah River Site F-Area

Figure H–4 shows the racial and ethnic composition of the minority population residing within 80 kilometers (50 miles) of F-Area at SRS projected to reside in the potentially affected area in the year 2010. In the interval between 1990 and 2010, the percentage of the total population composed of minorities is projected to increase from 37.9 percent to 42 percent. For comparison, during the 1990 census, minorities were found to compose approximately one-quarter of the total national population. By the year 2010, minorities are projected to compose closer to one-third of the total national population. The percentage of the minority population residing in the potentially affected area surrounding F-Area was larger than the corresponding national percentage in 1990, and is expected to remain so through the year 2010. Blacks are the largest minority group residing in the potentially affected area, while the Asian and Hispanic populations are projected to show the largest growth rates.

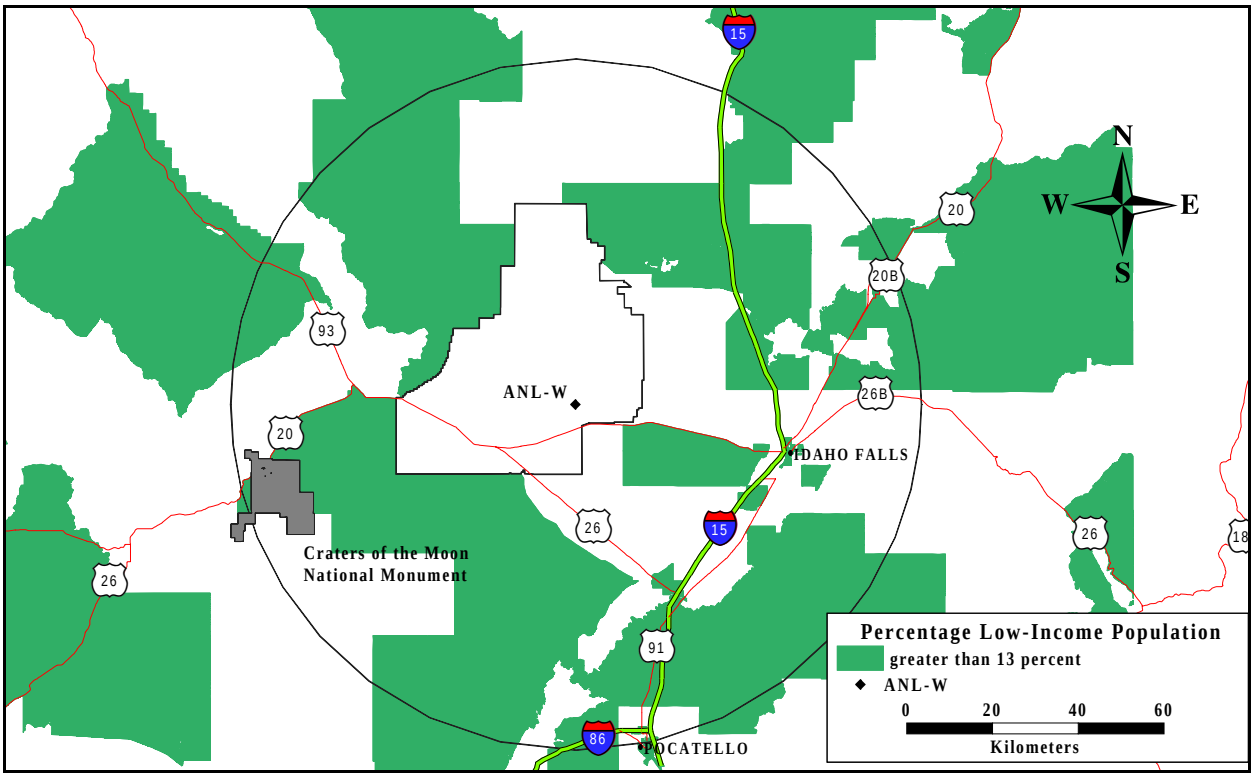

Figure H-3 Low-Income Population Residing Within 80 Kilometers (50 Miles) of ANL-W in 1990

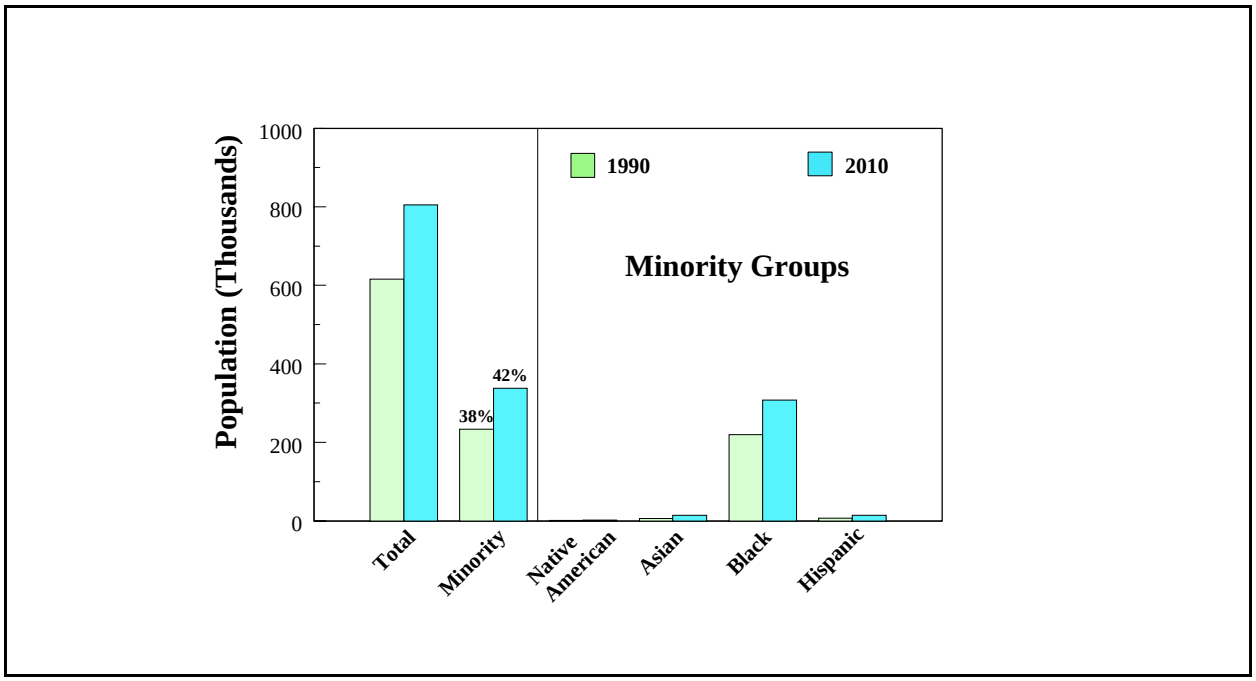

Figure H-4 Racial and Ethnic Composition of the Minority Population Residing Within 80 Kilometers (50 Miles) of the SRS F-Area in 2010

Figure H–5 shows the geographical distribution of minority populations residing near the SRS F-Area in 1990. Block groups for which the percentage of the minority population exceeds the national percentage are located throughout the potentially affected area.

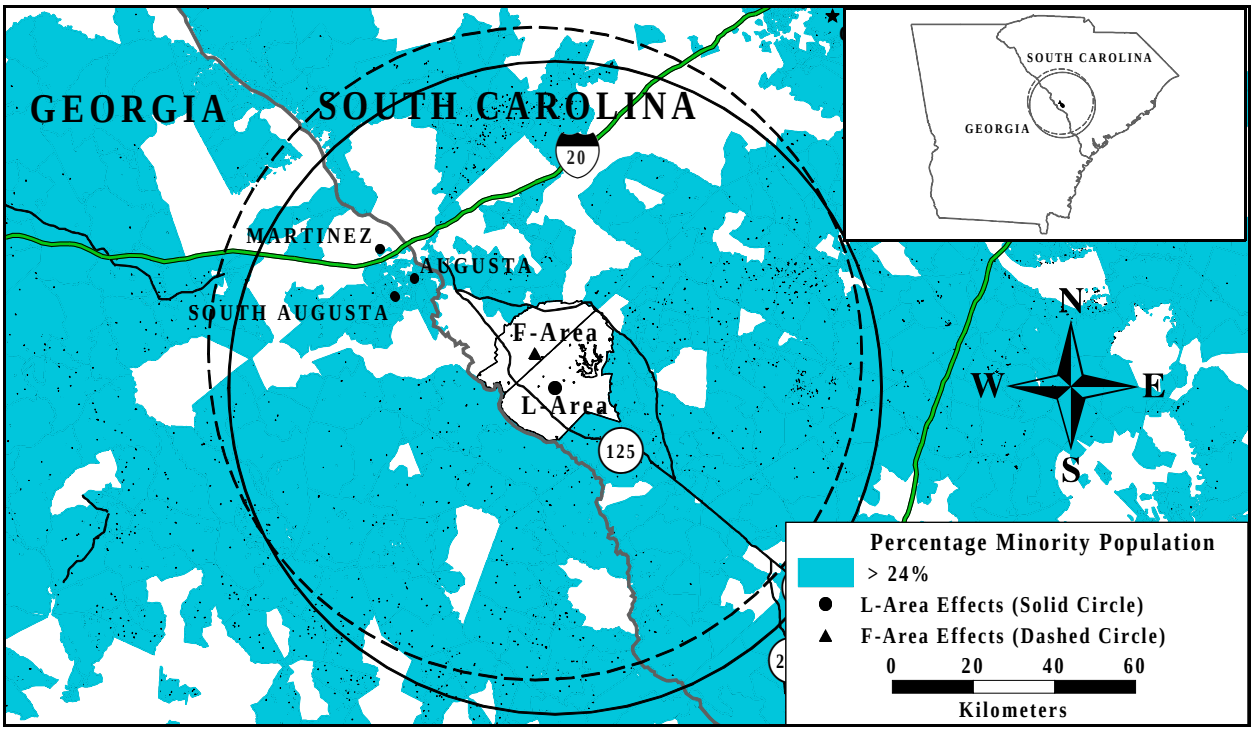

Figure H–5 Minority Population Residing Within 80 Kilometers (50 Miles) of SRS F-Area and L-Area in 1990

During the 1990 census, 18 percent of the residents within the potentially affected area surrounding F-Area reported incomes below the poverty threshold. Slightly over 13 percent of the national population reported incomes below the poverty threshold, and nearly 15 percent of the residents of the combined States of Georgia and South Carolina reported incomes below the poverty threshold during the same year. Thus, the percentage of low-income population residing within the potentially affected area exceeded that for the Nation and the States of Georgia and South Carolina. **Figure H–6** shows the geographical distribution of low-income residents surrounding the F-Area site. Block groups for which the percentage of low-income residents exceeds the corresponding national percentage are located throughout the potentially affected area.

As discussed in Chapter 4, implementation of the alternatives resulting in activities at F-Area would pose little risk to the public and the natural environment. Thus, potential activities for the treatment and management of sodium-bonded spent nuclear fuel at F-Area would not be expected to pose disproportionately high and adverse risks to potentially affected minority and low-income populations residing near F-Area.

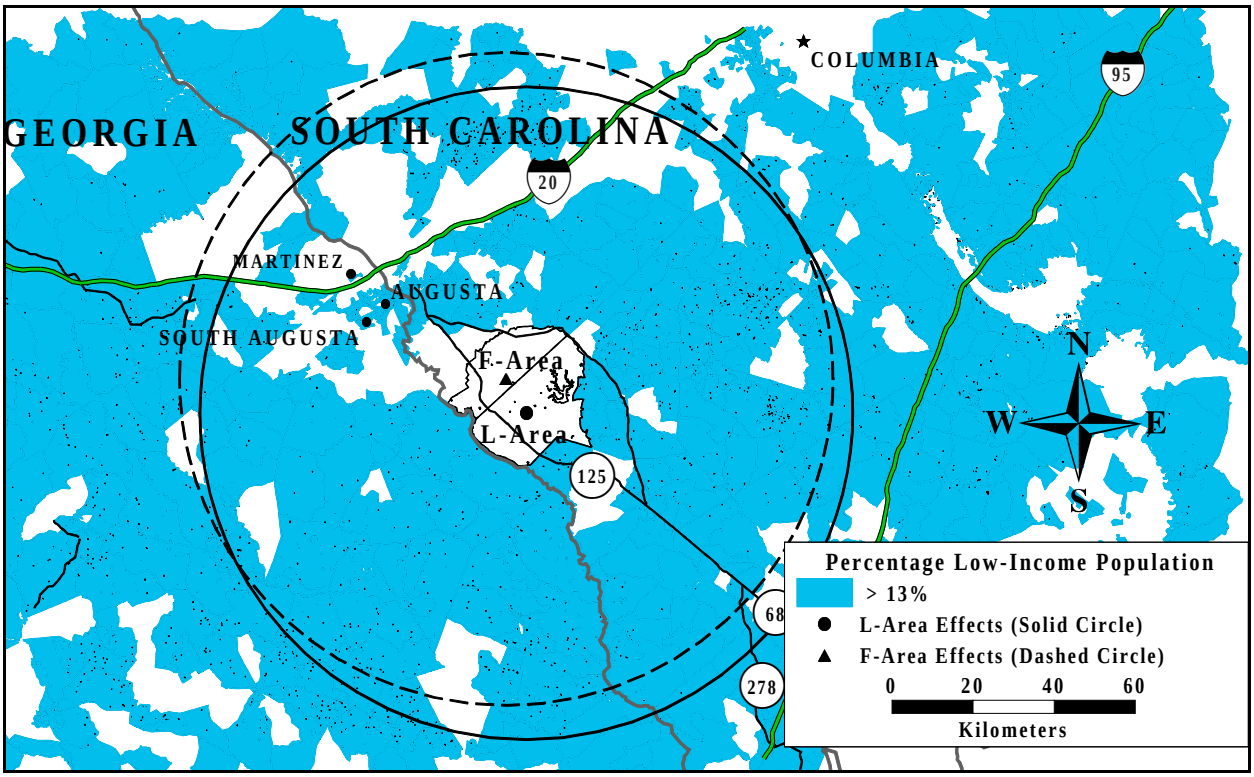

Figure H-6 Low-Income Populations Residing Within 80 Kilometers (50 Miles) of SRS F-Area and L-Area in 1990

H.5.3 The Savannah River Site L-Area

Figure H–7 shows the racial and ethnic composition of the minority population projected to reside in the potentially affected area surrounding the SRS L-Area by the year 2010. In the interval between 1990 and 2010, the percentage of the total population composed of minorities is projected to increase from 39.1 percent to 43 percent. For comparison, during the 1990 census, minorities were found to compose approximately one-quarter of the total national population. By the year 2010, minorities are projected to compose close to one-third of the total national population. The percentage of the minority population residing in the potentially affected area surrounding L-Area was larger than the corresponding national percentage in 1990, and is expected to remain so through the year 2010. Blacks are the largest minority group residing in the potentially affected area, while the Asian and Hispanic populations are projected to show the largest growth rates.

Figure H–5 shows the geographical distribution of minority populations residing near the SRS L-Area and F-Area. F-Area was discussed in Section H.5.2 above. As indicated in the figure, block groups for which the percentage of minority residents exceeds the national percentage are distributed throughout the potentially affected area surrounding L-Area.

During the 1990 census, 20.6 percent of the residents within the potentially affected area surrounding L-Area reported incomes below the poverty threshold. Slightly over 13 percent of the national population reported incomes below the poverty threshold, and nearly 15 percent of the residents of the combined States of Georgia and South Carolina reported incomes below the poverty threshold during the same year. Thus, the percentage low-income population residing within the potentially affected area exceeded that for the Nation and the States of Georgia and South Carolina. As shown in Figure H–6, block groups for which the percentage of low-income residents exceeds the corresponding national percentage are located throughout the potentially affected area.

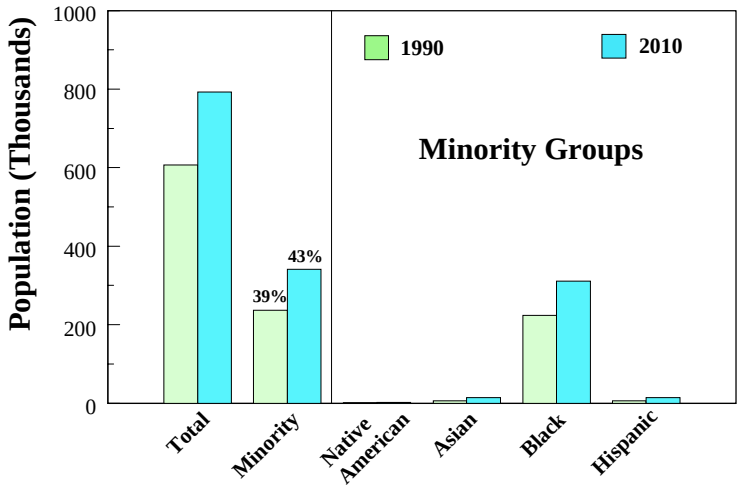

Figure H-7 Racial and Ethnic Composition of the Minority Population Residing Within 80 Kilometers (50 Miles) of the SRS L-Area in 2010

As discussed in Chapter 4, implementation of the alternatives resulting in activities at L-Area would pose little risk to the public and the natural environment. Thus, potential activities for the treatment and management of sodium-bonded spent nuclear fuel at L-Area would not be expected to pose disproportionately high and adverse risks to potentially affected minority and low-income populations residing near L-Area.

H.6 Results for Transportation Routes

As discussed in Chapter 4 of this EIS, no significant radiological or nonradiological risks along representative transportation routes would result from implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel. Therefore, implementation of these alternatives would pose no disproportionately high and adverse risks to minority and low-income groups within the general public.

H.7 Other Environmental Impacts

No significant adverse impacts to biotic resources, air resources, socioeconomics, land use, or cultural resources were identified in Chapter 4. Therefore, no disproportionately high and adverse impacts were identified for any segment of the population. None of the alternatives would have a significant adverse impact on the previously mentioned resources because, under all of the alternatives, all activities associated with the treatment and management of sodium-bonded spent nuclear fuel would take place within existing facilities at ANL-W and SRS.

H.8 Cumulative Impacts

Based on the analysis of the environmental impacts evaluated in this EIS, along with the impacts of other past, present, and reasonably foreseeable future activities, no reasonably foreseeable cumulative adverse impacts are expected to affect the surrounding minority and low-income populations.

H.9 REFERENCES

CEQ (Council on Environmental Quality), 1997, *Environmental Justice Guidance Under the National Environmental Policy Act*, Executive Office of the President, Washington, DC, December 10.

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APPENDIX I
ECOLOGICAL RESOURCES

I.1 INTRODUCTION

Table I–1 contains a listing of the scientific names of animal and plant species found in the text. Species are grouped and listed in alphabetical order by common name.

Table I–1 Scientific Names of Animal and Plant Species Referred to in the Text

Common Name	*Scientific Name*
Mammals	
Black-tailed jackrabbit	*Lepus californicus*
Coyote	*Canis latrans*
Eastern cottontail	*Sylvilagus floridanus*
Elk	*Cervus elaphus*
Feral hog	*Sus scrofa*
Gray fox	*Urocyon cinereoargenteus*
Gray wolf	*Canis lupus*
Mountain lion	*Felis concolor*
Mule deer	*Odocoileus hemionus*
Pronghorn	*Antilocapra americana*
Pigmy rabbit	*Brachylagus idahoensis*
Raccoon	*Procyon lotor*
Townsend's big-eared bat	*Corynorhinus townsendii*
Townsend's ground squirrel	*Spermophilus townsendii*
Whitetail deer	*Odocoileus virginianus*
Birds	
Bald eagle	*Haliaeetus leucocephalus*
Black vulture	*Coragyps atratus*
Carolina chickadee	*Parus carolinensis*
Common crow	*Corvus brachyrhynchos*
Cooper's hawk	*Accipiter cooperi*
Golden eagle	*Aquila chrysaetos*
Loggerhead shrike	*Lanius ludovicianus*
Peregrine falcon	*Falco peregrinus*
Prairie falcon	*Falco mexicanus*
Red-cockaded woodpecker	*Picoides borealis*
Sage grouse	*Centrocercus urophasianus*
Sage sparrow	*Amphispiza belli*
Wood stork	*Mycteria americana*
Reptiles	
American crocodile	*Crocodylus acutus*
American alligator	*Alligator mississippiensis*
Eastern box turtle	*Terrapene carolina*
Gopher snake	*Pituophis melanoleucus*
Short-horned lizard	*Phyrnosoma douglassi*
Amphibians	
Slimy salamander	*Plethodon glutinosus*

Common Name	Scientific Name
Fish	
American shad	*Alosa sapidissima*
Black crappie	*Pomoxis nigromaculatus*
Blueback herring	*Alosa aestivalis*
Bluegill	*Lepomis macrochirus*
Brook trout	*Salvelinus fontinalis*
Creek chubsucker	*Erimyzon oblongus*
Dusky shiner	*Notropis cummingsae*
Kokanee salmon	*Oncorhynchus nerka*
Lake chubsucker	*Erimyzon sucetta*
Largemouth bass	*Micropterus salmoides*
Mosquitofish	*Gambusia affinis*
Mountain whitefish	*Prosopium williamsoni*
Mud sunfish	*Acantharchus pomotis*
Rainbow trout	*Oncorhynchus mykiss*
Redbreast sunfish	*Lepomis auritus*
Redfin pickerel	*Esox americanus*
Shorthead sculpin	*Cottus confusus*
Speckled dace	*Rhinichthys osculus*
Spotted sunfish	*Lepomis punctatus*
Striped bass	*Morone saxatilis*
Sunfish	*Lepomis spp.*
Threadfin shad	*Dorosoma petenense*
Yellow bullhead	*Ictalurus natalis*
Yellowfin shiner	*Notropis lutipinnis*
Mollusks	
Giant oyster	*Crassostrea gigantissima*
Plants	
American ginseng	*Panax quinquefolium*
Bald cypress	*Taxodium distichum*
Big sagebrush	*Artemisia tridentata*
Bluebunch wheatgrass	*Agropyron spicatum*
Bottlebrush squirreltail	*Sitanion hystrix*
Button snakeroot	*Erynglum yuccifolium*
Cottonwood	*Populus spp.*
Crested wheatgrass	*Agropyron desertorum*
Cypress	*Taxodium spp.*
Giant wildrye	*Elymus condensatus*
Gray horsebrush	*Tetradymia canescens*
Green rabbitbrush	*Chrysothamnus greenei*
Hickory	*Carya spp.*
Indian ricegrass	*Oryzopsis hymenoides*
Juniper	*Juniperus spp.*
Loblolly pine	*Pinus taeda*
Longleaf pine	*Pinus palustris*
Low sagebrush	*Artemisia arbuscula*
Needle-and-tread grass	*Stipa comata*
Oak	*Quercus spp.*
Oconee azalea	*Rhododendron flammeum*
Poverty-weed	*Monolepis nuttaliana*
Prickly pear cactus	*Opuntia spp.*

Common Name	***Scientific Name***
Rabbitbrush	*Chrysothamnus spp.*
Redroot	*Lachnanthese carolinianum*
Rush	*Juncus spp.*
Sagebrush	*Artemisia spp.*
Saltbush	*Atriplex spp.*
Slash pine	*Pinus elliottii*
Smooth purple coneflower	*Echinacea laevigata*
Thickspike wheatgrass	*Agropyron dasytachym*
Threetip sagebrush	*Artemisia tripartita*
Tupelo	*Nyssa slyvotica*
Utah juniper	*Juniperus osteosperma*
Western wheatgrass	*Agropyron smithii*
Willow	*Salix spp.*
Winterfat	*Eurotia lanata*

DEPARTMENT OF ENERGY

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, Idaho National Engineering and Environmental Laboratory, Idaho

AGENCY: U.S. Department of Energy.

ACTION: Notice of intent to prepare an environmental impact statement.

SUMMARY: The Department of Energy announces its intent to prepare an Environmental Impact Statement (EIS) pursuant to the National Environmental Policy Act (NEPA) for the proposed electrometallurgical treatment of Department of Energy-owned sodium-bonded spent nuclear fuel in the Fuel Conditioning Facility at Argonne National Laboratory-West (ANL-W). ANL-W, a center of nuclear technology development and testing, is located on the Idaho National Engineering and Environmental Laboratory (INEEL) site in southeastern Idaho. The Department proposes to treat its inventory of sodium-bonded spent nuclear fuel to remove and stabilize the reactive metallic sodium constituent and to produce metal and ceramic waste forms, considered to be high-level waste, that would facilitate interim storage and ultimate disposal of this material. The EIS will evaluate reasonable action alternatives to electrometallurgical treatment in the Fuel Conditioning Facility at ANL-W and a no-action alternative. The Department invites the general public, other Federal agencies, American Indian tribes, state and local governments, and all other interested

parties to comment on the scope of this EIS.

DATES: To ensure consideration in the preparation of the draft EIS, comments should be transmitted or postmarked by April 8, 1999. Comments submitted after that date will be considered to the extent practicable.

The Department will conduct public scoping meetings in Idaho Falls and Boise in Idaho, near the Department's Savannah River Site (SRS) in South Carolina, and in the Washington, DC area, to provide the public with information about the proposed project and to receive oral and written comments on the scope of the EIS, including reasonable alternatives and environmental issues that the Department should consider. The dates, times, and locations for these public meetings are as follows:

March 9, 1999 (6:00 pm–9:00 pm)
Shilo Inn, 780 Lindsay Blvd., Idaho Falls, ID 83402, (208) 523–0088

March 11, 1999 (6:00 pm–9:00 pm)
Boise Centre on the Grove, 850 West Front Street, Boise, ID 83702, (208) 336–8900

March 15, 1999 (6:00 pm–9:00 pm)
North Augusta Community Center, 495 Brookside Avenue, North Augusta, SC 29842, (803) 441–4290

March 18, 1999 (2:00 pm–5:00 pm)
Hyatt Regency Crystal City, 2799 Jefferson Davis Highway, Arlington, VA 22202, (703) 418–1234

These public scoping meetings will also be announced in local media at least 15 days prior to the meeting dates. During the first hour of each meeting attendees may register, view displays and discuss issues and concerns informally with Department representatives, after which there will be a formal presentation, a follow-on question, answer, and comment period, and the opportunity for additional informal discussions.

ADDRESSES: Written comments on the scope of the EIS, requests to speak at the public scoping meetings, requests for special arrangements to enable participation at scoping meetings (e.g., an interpreter for the hearing impaired), requests to be placed on the EIS document distribution list, and questions concerning the project should be sent to: Susan Lesica, Document Manager, Office of Nuclear Facilities Management, Office of Nuclear Energy, Science, and Technology, U.S. Department of Energy, NE–40, 19901 Germantown Road, Germantown, Maryland 20874–1290

Interested parties may also submit comments and requests by facsimile to (877) 621–8288, or they may call (877) 450–6904 to leave a detailed message with their comments and requests. These are both toll-free telephone numbers. Comments and requests may also be submitted by electronic mail to emtEIS@hq.doe.gov.

FOR FURTHER INFORMATION CONTACT: For general information on the Department of Energy NEPA process, please contact: Carol Borgstrom, Director, Office of NEPA Policy and Assistance, Office of Environment, Safety and Health, U.S. Department of Energy, EH–42, 1000 Independence Avenue, SW, Washington, DC 20585–0119, 202–586–4600 or leave a message at 1–800–472–2756.

SUPPLEMENTARY INFORMATION:

Background

The Department of Energy is responsible for the safe and efficient management of 250 different types of spent nuclear fuel, including its ultimate disposition (which is expected to be disposal in a geologic repository). Some Department spent fuels may be suitable for disposal with little or no stabilizing treatment. Other spent fuel types may not be suitable for disposal without significant treatment or stabilization.

One type of spent nuclear fuel that may not be suitable for disposal without treatment is sodium-bonded spent nuclear fuel. Sodium-bonded spent nuclear fuel contains metallic sodium, a highly reactive material. Metallic sodium reacts vigorously with water or moist air producing heat, potentially explosive hydrogen gas, and sodium hydroxide, a corrosive substance. Sodium metal was used as a heat transfer medium within the stainless steel cladding of sodium-bonded fuel and as coolant in the nuclear reactors in which these fuels were used. To the extent possible, the highly reactive sodium has been removed from external surfaces of these fuels after their use, but a portion remains bonded to the uranium metal alloy fuel within the cladding and cannot be removed without further treatment. The presence of reactive or pyrophoric material, such as metallic sodium, could complicate the process of qualifying and licensing such spent fuel for disposal, which would require data and predictive analyses sufficient to demonstrate that emplacement of the spent fuel would not adversely affect a repository's ability to protect the environment and public health.

The Department believes that treatment to remove metallic sodium and convert this spent nuclear fuel into a compact waste form would reduce complications of disposal qualification and licensing. Technologies for spent nuclear fuel treatment that might facilitate such qualification and licensing should therefore be considered in reaching a disposition decision for Department-owned sodium-bonded fuels. One such technology for sodium-bonded spent fuel disposition is the electrometallurgical treatment technique that the Department is developing and demonstrating at the Argonne National Laboratory. This technology is currently the most developed for treatment of sodium-bonded spent fuel. In addition to electrometallurgical treatment, the Department will examine all reasonable alternative technologies and assess the technical risks associated with these various potential solutions.

In a 1995 report, the National Research Council Committee on Electrometallurgical Techniques for DOE Spent Fuel Treatment recommended that the Department confirm the technical feasibility and cost effectiveness of electrometallurgical treatment of its sodium-bonded spent nuclear fuel through a technology demonstration using sodium-bonded spent nuclear fuel that had been removed from the Experimental Breeder Reactor–II (EBR–II) at ANL–W. Prior to acting on the recommendation, the Department prepared the Environmental Assessment for the Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West (DOE/EA–1148) and issued a Finding of No Significant Impact on May 15, 1996. The demonstration project addresses both kinds of spent fuel assemblies in the EBR–II spent nuclear fuel inventory. These are driver fuel assemblies and blanket fuel assemblies, and they total about 26 metric tons of heavy metal (MTHM).

One MTHM is equal to 2,200 pounds of uranium, thorium, or plutonium. The driver fuel contains highly enriched uranium and was used in the active region of the nuclear reactor core. Blanket fuel contains depleted uranium and was used in areas around and near the driver fuel in the reactor core. The demonstration project now nearing completion involves treatment of 100 EBR–II driver assemblies and 25 EBR–II blanket assemblies (approximately 1.6 MTHM, or only 6.25% of the EBR–II inventory) in the Fuel Conditioning Facility at ANL–W. The research and demonstration project was initiated in June 1996 and is scheduled to be completed in August 1999.

The National Research Council is continuing to evaluate the electrometallurgical treatment research

and demonstration project. In its most recent report titled, Electrometallurgical Techniques for U.S. Department of Energy Spent Fuel Treatment—Spring 1998 Status Report on Argonne National Laboratory's R&D Activity (National Academy Press, Washington, DC, 1998), the Council acknowledged progress in the demonstration and recommended that the demonstration be carried to completion. The Department believes that this progress and the absence of significant roadblocks to successful completion of the demonstration warrant proposing electrometallurgical treatment of the remainder of the EBR-II and other sodium-bonded spent fuels (i.e., a total of 62 MTHM) and is initiating the environmental review process under NEPA. Accordingly, the Department is announcing its intent to prepare an EIS for the proposed treatment of the remainder of Department sodium-bonded spent nuclear fuel.

Data from the ongoing demonstration project will be used in preparing the EIS. The National Research Council will issue a final report on the technology demonstration upon completion of the demonstration project. DOE will consider the Council's report in reaching a decision regarding the disposition of sodium-bonded spent nuclear fuel.

Purpose and Need for Agency Action

In a 1995 agreement with the State of Idaho [Settlement Agreement and Consent Order issued by the Court on October 17, 1995, in the actions *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.), the Department committed to remove all spent nuclear fuel from Idaho by 2035. More than 98 percent of the Department's sodium-bonded spent nuclear fuel is located at INEEL near Idaho Falls, Idaho, and is subject to the requirements of the Settlement Agreement and Consent Order. The remaining Department sodium-bonded spent nuclear fuel included in the proposed action is at the Hanford Reservation in Richland, Washington, the Sandia National Laboratories in Albuquerque, New Mexico, and the Oak Ridge National Laboratory in Oak Ridge, Tennessee. In order to remove sodium-bonded spent nuclear fuel from the State of Idaho to meet the terms of the Settlement Agreement and Consent Order referenced above, the Department believes the best approach would be to stabilize or remove the reactive metallic sodium constituent and prepare a waste form that may be more assuredly demonstrated to be acceptable for disposal.

It is also prudent to evaluate the electrometallurgical treatment proposal and alternative technologies now, while the Department is performing site characterization activities for a potential geologic repository. Contemplated waste forms resulting from treatment or packaging of sodium-bonded spent fuel should be developed as much as possible in parallel with any repository development to promote consistency between the two efforts and to minimize technical risks associated with waste form qualification and acceptance for geologic disposal. While the alternative technologies for treatment of sodium-bonded spent fuel may not be as mature as the electrometallurgical treatment technology, their potential utility can be assessed in this EIS. Should the Department decide, after completing this EIS, to pursue a disposition path other than electrometallurgical treatment, there will still be sufficient time to develop an alternative technology. If a treatment technology decision is significantly delayed, however, the Department could functionally lose its expertise and corporate experience in the specialized electrometallurgical treatment technology at ANL-West, which would hamper future consideration and increase the cost of electrometallurgical treatment for sodium-bonded spent fuel disposal. Therefore, the Department believes it is prudent to proceed now with this EIS for electrometallurgical treatment of sodium-bonded spent fuel.

Proposed Action

The Department proposes to treat its sodium-bonded spent nuclear fuel [1] using the electrometallurgical treatment process in the Fuel Conditioning Facility at ANL-W. Electrometallurgical processing involves the dissolution of spent nuclear fuel by use of an electric current in a molten salt mixture. The uranium in the fuel would be collected from a molten salt mixture at the cathode and subsequently melted and cast into metal ingots. The metal cladding from the fuel elements and noble metal fission products would be retrieved undissolved from the anode, melted, and cast into metal ingots. Remaining fission products and all transuranic elements would be removed from the molten salt mixture by ion exchange and subsequently isolated in a ceramic waste form. In this process, the metallic sodium in the spent nuclear fuel would be converted to non-reactive sodium chloride (same composition as table salt) and incorporated in the ceramic waste form.

Based on available information, the Department believes the electrometallurgical treatment process would produce metal and ceramic high-level radioactive waste forms that could be qualified and licensed for disposal. In addition, uranium would be separated from both the driver fuel and the blanket fuel and not disposed of. The highly enriched uranium separated from the driver fuel assemblies would be immediately blended down in the Fuel Conditioning Facility to form low-enriched uranium. This low-enriched uranium and the depleted uranium that would be separated from blanket fuel assemblies would be cast as metal ingots and stored with other uranium metal inventories at INEEL. The disposition of these materials would be included in future Departmental decisions regarding other similar materials.

The sodium-bonded spent nuclear fuel inventory being proposed for electrometallurgical treatment totals approximately 62 MTHM. This inventory of sodium-bonded spent nuclear fuel is currently stored as follows:

- Approximately 24 MTHM of EBR-II sodium-bonded driver and blanket assemblies currently stored at ANL-W and approximately 2 MTHM at the Idaho Nuclear Technology and Engineering Center (INTEC), both located at INEEL.
- Approximately 35 MTHM of sodium-bonded spent nuclear fuel from the Fermi-1 reactor, currently stored at INTEC.
- Less than one MTHM consisting of six irradiated sodium-bonded fuel assemblies and a number of sodium-bonded spent nuclear fuel pins currently stored at the Hanford Reservation near Richland, Washington.
- Less than 0.1 MTHM consisting of experimental capsules currently stored at INTEC and Clinch River Breeder Reactor Program experimental capsules currently stored at Sandia National Laboratories, Albuquerque, New Mexico.
- Less than 0.01 MTHM consisting of miscellaneous fast reactor development fuel currently stored at Oak Ridge National Laboratory, Oak Ridge, Tennessee.

[1] The Department has no plan or intention to apply this technology to any other types of spent nuclear fuel. Nevertheless, the Department can foresee a potential need to treat small quantities of certain spent fuels if a non-treatment (e.g., high integrity can) approach to disposing of such spent fuels were to be determined not to meet disposal requirements. In that case, electrometallurgical treatment might be among the reasonable alternative treatment technologies that would be considered.

The sodium-bonded spent nuclear fuels located at the Hanford Reservation, Oak Ridge, and Sandia can be transported to INEEL pursuant to the Record of Decision (60 FR 28680, June 1, 1995) for the Department of Energy's Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Final Environmental Impact Statement (DOE/EIS–0203–F), under the Settlement Agreement and Consent Order described above. These spent fuels pose the same waste form acceptability issues and are amenable to the same treatments as the EBR–II and Fermi-1 fuels stored at INEEL.

Alternatives To Be Evaluated

The Department has identified the following alternatives to the proposed electrometallurgical treatment of sodium-bonded spent nuclear fuel in the Fuel Conditioning Facility at ANL–W.

A. *No Action Alternative:* Under this alternative, the Department would not treat its sodium-bonded spent nuclear fuel to facilitate disposal. Analyses will address the viability of disposal without treatment, and the impacts of continued storage at current locations. Both temporary storage (to await alternative technology development) and indefinite storage (in lieu of disposal) will be considered in these analyses. Indefinite storage of spent nuclear fuel in Idaho would not be consistent with the Settlement Agreement and Consent Order in which the Department committed to remove all spent nuclear fuel from Idaho by 2035.

B. *Technology Alternatives:* The National Research Council independently assessed other treatment technologies as possible alternatives to electrometallurgical treatment for EBR–II sodium-bonded spent nuclear fuel. It concluded that all of the alternative treatment processes evaluated, except the Plutonium-Uranium Extraction (PUREX) process, are at an early stage of development. Significant research, development, and demonstrations would be required to develop these alternative treatment processes to the level of technical maturity of the electrometallurgical treatment process for sodium-bonded spent fuel. However, the Department will examine and analyze these alternative technologies:

1. PUREX Process. This solvent extraction method for separating and purifying uranium, plutonium, and other radionuclides from spent nuclear fuel and irradiated targets is presently practiced at the SRS for stabilization of materials that are not suitable for prolonged storage in their present forms, and as such pose potential health and safety risks. In the Savannah River Site Spent Nuclear Fuel Management EIS, the Department is currently evaluating use of the PUREX process for stabilizing approximately 17 MTHM of previously declad EBR–II spent nuclear fuel stored at the SRS site. Use of the PUREX facility to treat sodium-bonded spent nuclear fuel being considered under this alternative would require development of specific processes for removing the stainless-steel cladding and sodium from the spent fuel.

The Department intends to evaluate the PUREX process at SRS as an alternative to electrometallurgical treatment of the sodium-bonded spent fuel inventory. Material streams from the PUREX process would be uranium trioxide, plutonium metal, high-level waste in the form of borosilicate glass canisters, and grouted low-level waste.

2. High-Integrity Cans. Under this alternative, the spent fuel would be placed in high-integrity cans, after as little treatment as necessary, to prepare it for disposal. This alternative would include removal of as much of the metallic sodium as possible from the spent fuel prior to loading it in the cans.

3. Glass Material Oxidation and Dissolution System (GMODS). The basic concept is to combine unprocessed sodium-bonded spent nuclear fuel and a sacrificial oxide, lead-borate glass, in a glass melter at a temperature of 800–1000 °C. The uranium and the plutonium in the spent fuel would be converted into oxides and dissolved in the glass. Options to be analyzed are direct production of a borosilicate glass waste form from the melt, using the melt as a feed to the PUREX process, and coupling GMODS to the SRS Defense Waste Processing Facility, where the melt would be fed directly to the existing glass melter. 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.

4. Melt and Dilute Process. The process would be similar to that proposed for the treatment of aluminum-based spent nuclear fuels at the SRS. The sodium-bonded spent fuel would be chopped and melted at approximately 650 to 850 °C and then diluted by the addition of depleted uranium and iron.

5. Chloride Volatility Process. This process would use the differences in volatilities of chloride compounds to separate the constituents of spent nuclear fuel. The major steps are: (1) high-temperature chlorination at about 1500 °C and conversion of metallic fuel and cladding to gaseous chloride compounds; (2) removal of the transuranic chlorides and most of the fission products in a molten zinc chloride bed at approximately 400 °C; (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 would then be converted into either fluorides or oxides for disposal.

6. Direct Plasma Arc-Vitreous Ceramic Process. In this process, the spent nuclear fuel would be melted and oxidized with the help of an oxygen lance in a rotating furnace containing molten ceramic materials at a temperature of 1600 °C 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. When the spent 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 a canister.

C. *Location Alternatives:* An alternative location for electrometallurgical treatment on the INEEL site is the Test Area North Hot Cell Facility. This alternative to the Fuel Conditioning Facility at ANL–W will be evaluated in the EIS.

U.S. Nonproliferation Policy Implications

The United States does not encourage the civil use of plutonium, and accordingly, does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes. Consistent with this policy, the proposed action would not separate plutonium from the processed sodium-bonded spent fuels. Further, by removing and diluting the highly enriched uranium in the sodium-bonded driver fuel to low-enriched uranium, the proposed project would support the U.S. goal of minimizing civilian use of highly enriched uranium. However, to address the concerns that the treatment of this fuel could encourage reprocessing in other countries, the Department (Office of Nonproliferation and National Security) will assess the nonproliferation impacts of all the treatment technologies in the draft EIS. This assessment will be made publicly available during the EIS process. The combination of the information contained in the draft EIS, the public comment in response to the draft EIS, and the nonproliferation impacts assessment report will enable

the Department to make a sound decision regarding how to manage the sodium-bonded spent nuclear fuel.

Preliminary Identification of Environmental Issues

The issues listed below have been tentatively identified for analysis in the EIS. This list is presented to facilitate public comment on the scope of the EIS. It is not intended to be all-inclusive or to predetermine the potential impacts of any of the alternatives. The Department seeks public comment on the adequacy and inclusiveness of the following issues.

- Potential impact on ecosystems, including air quality, surface, and groundwater quality, and plants and animals.
- Potential health and safety impact to on-site workers and to the public resulting from operations, including reasonably foreseeable accidents.
- Potential health and safety, environmental, and other impact related to the transport of spent nuclear fuel for treatment.
- Considerations related to the generation, treatment, storage, and disposal of wastes, including the potential acceptability of waste forms at a geologic repository.
- Potential cumulative impacts of electrometallurgical and alternative treatment process operations, including relevant impact from other past, present, and reasonably foreseeable activities at the operation site.
- Potential impact on cultural resources.
- Potential socioeconomic impact, including any disproportionate impacts on minority and low income populations.
- Pollution prevention and waste minimization opportunities.

Related NEPA Documentation

NEPA documents that have been or are being prepared for activities related to the proposed action include, but are not limited to, the following:

- U.S. Department of Energy, "Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West; Environmental Assessment," DOE/EA-1148, May 1996
- U.S. Department of Energy, "Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management; Final Environmental Impact Statement," DOE/EIS-0203-F, April 1995, and Record of Decision, May 30, 1995
- U.S. Department of Energy, "Savannah River Site, Spent Nuclear Fuel Management, Draft Environmental Impact Statement," DOE/EIS-0279D, December 1998
- U.S. Department of Energy, "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," DOE/EIS-0250—in preparation

Public Involvement Opportunities

The Department encourages public involvement in the preparation of the EIS and solicits public comments on its scope and content, as well as public participation at the public scoping meetings in Idaho, South Carolina, and the Washington, DC area. Department of Energy personnel will be available at the scoping meetings to explain the proposed project and answer questions. The Department will designate a facilitator for the scoping meetings. At the opening of each meeting, the facilitator will establish the order of speakers and will announce any additional procedures necessary for conducting the meeting. Additionally, during the first hour of each meeting attendees may register, view displays and discuss issues and concerns informally with Department representatives, after which there will be a formal presentation, a question and answer, and comment period, and the opportunity for additional informal discussions. To ensure that all persons wishing to make a presentation during the period for questions and answers or comments are given the opportunity to speak, a five-minute limit may be applied for each speaker, except that public officials and representatives of groups would be allotted ten minutes each. The Department encourages those providing oral comments to also submit them in writing. Comment cards will be available at the meetings for those who prefer to submit their comments in writing. Speakers may be asked clarifying questions to ensure that the Department representatives fully understand the comments and suggestions made by meeting participants, but the scoping meetings will not be conducted as evidentiary hearings.

The Department will make transcripts of public scoping meetings, copies of background documents, and other materials related to the proposed project and the development of the EIS available for public review in the following reading rooms:

Washington, DC: U.S. Department of Energy, Freedom of Information Reading Room, Forrestal Building, Room 1E-190, 1000 Independence Avenue, SW, Washington, DC 20585-0117, 202-586-3142

Idaho Falls, Idaho: Idaho National Engineering and Environmental Laboratory, DOE—Idaho Operations Office Public Reading Room, 1776 Science Center Drive, Idaho Falls, ID 83415, 208-526-0271

Richland, Washington: [for vicinity of the Hanford Reservation], DOE Public Reading Room, 2770 University Drive, CIC, Room 101L, Richland, WA 99352, 509-372-7443, (Fax) 509-372-7444

Albuquerque, New Mexico: [for vicinity of Sandia National Laboratories], University of New Mexico, Government Information Department, Zimmerman Library, Albuquerque, NM 87131-1466, 505-277-0582

Aiken, South Carolina: [for vicinity of the Savannah River Site], University of South Carolina—Aiken, Gregg-Graniteville Library, 171 University Parkway, Aiken, SC 29803, 803-648-6851

Oak Ridge, Tennessee: [for vicinity of the Oak Ridge National Laboratory], DOE Public Reading Room, 230 Warehouse Road, Bldg 1916-T-2, Suite 300, Oak Ridge, TN 37831, 423-241-4780 and DOE Information Resource Center, 105 Broadway Avenue, Oak Ridge, TN 37830, 423-241-4582

NEPA Process

The EIS for Electrometallurgical Treatment of Sodium-Bonded Spent Nuclear Fuel in the Fuel Conditioning Facility at ANL-W will be prepared in accordance with the NEPA of 1969, the Council on Environmental Quality's Regulations for Implementing the Procedural Provisions of NEPA (40 CFR Parts 1500-1508), and the U.S. Department of Energy NEPA Implementing Procedures (10 CFR Part 1021).

A 45-day comment period on the draft EIS is planned, during which public hearings to receive comments will be held. The draft EIS is scheduled to be issued in July 1999. Availability of the draft EIS, the dates of the public comment period, and information about the public hearings will be announced in the **Federal Register** and in local news media when the draft EIS is distributed. The final EIS, which will consider and respond to the public comments received on the draft EIS, is scheduled to be issued in December 1999. No sooner than 30 days after the U.S. Environmental Protection Agency's notice of availability of the final EIS is published in the **Federal Register**, the Department will issue its Record of

Decision and publish it in the **Federal Register**.

Signed in Washington, DC, this 16th day of February 1999.

Peter N. Brush,

Principal Deputy Assistant Secretary, Environment, Safety and Health.

[FR Doc. 99-4289 Filed 2-19-99; 8:45 am]

BILLING CODE 6450-01-P

NEPA DISCLOSURE STATEMENT FOR PREPARATION OF EIS FOR THE TREATMENT AND MANAGEMENT OF SODIUM-BONDED SPENT NUCLEAR FUEL

CEQ regulations at 40 CFR 1506.5(c), which have been adopted by DOE (10 CFR 1021), require contractors who will prepare an EIS to execute a disclosure specifying that they have no financial or other interest in the outcome of the project. The term "financial interest or other interest in the outcome of the project," for the purposes of this disclosure, is defined in the March 23, 1981 guidance "Forty Most Asked Questions Concerning CEQ's National Environmental Policy Act Regulations," 46 FR 18026-18038 at Question 17a and b.

"Financial or other interest in the outcome of the project 'includes' any financial benefit such as a promise of future construction or design work in the project, as well as indirect benefits the contractor is aware of (e.g., if the project would aid proposals sponsored by the firm's other clients)." 46 FR 18026-18038 at 18031.

In accordance with these requirements, the offeror and any proposed subcontractors hereby certify as follows: (check either (a) or (b) to assure consideration of your proposal)

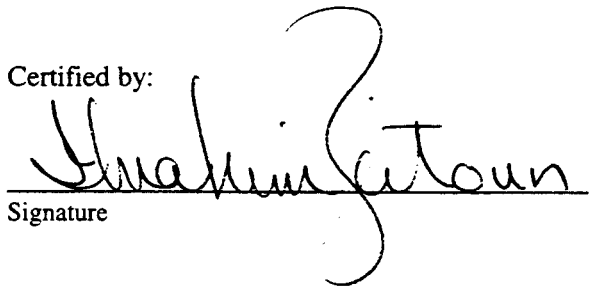
(a) ✓ Offeror and any proposed subcontractor have no financial interest in the outcome of the project.

(b) _____ Offeror and any proposed subcontractor have the following financial or other interest in the outcome of the project and hereby agree to divest themselves of such interest prior to award of this contract.

Financial or Other Interests:

1.
2.
3.

Certified by:



Signature

Ibrahim H. Zeitoun, Ph.D.
Name

Corporate Vice President
Integrated Environmental Services Operation

5/25/99
Date

Science Applications International Corporation