From: Dozier, Tami
Sent: Thursday, May 20, 2021 5:13 PM
To: AdvancedReactors-GEISDocsPEm Resource
Subject: Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel - Summary. DOE/EIS-0306
Attachments: DOE-EIS-0306-Summary-2000.pdf

Hearing Identifier: AdvancedReactors_GEIS_Public
Email Number: 26

Mail Envelope Properties (BY3PR09MB7585FEE15A0D6ADF8AA60E03F42A9)

Subject: Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel - Summary. DOE/EIS-0306
Sent Date: 5/20/2021 5:13:22 PM
Received Date: 5/20/2021 5:13:24 PM
From: Dozier, Tami

Created By: Tamsen.Dozier@nrc.gov

Recipients:
"AdvancedReactors-GEISDocsPEm Resource"
<AdvancedReactors-GEISDocsNPR.Resource3@nrc.gov>
Tracking Status: None

Post Office: BY3PR09MB7585.namprd09.prod.outlook.com

Files	**Size**	**Date & Time**
MESSAGE	3	5/20/2021 5:13:24 PM
DOE-EIS-0306-Summary-2000.pdf	1408929	

Options
Priority: Normal
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:

Department of Energy
Washington, DC 20585

July 14, 2000

Dear Interested Party:

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

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

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

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

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

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

Sincerely,

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

Enclosure

DOE/EIS - 0306
July 2000

Final Environmental Impact Statement

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

Summary

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

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

AVAILABILITY OF THE DRAFT SBSNF EIS

General questions regarding this EIS or for a copy of this EIS, please contact:

Susan M. Lesica, Document Manager
Office of Nuclear Facilities Management (NE-40)
Office of Nuclear Energy, Science, and Technology
U.S. Department of Energy
19901 Germantown Road
Germantown, MD 20874
Attention: SBSNF EIS
Telephone: (301) 903-8755

Printed with soy ink on recycled paper

COVER SHEET

Responsible Agency: United States Department of Energy (DOE)

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

Locations: Idaho, South Carolina

Contact: For copies of this environmental impact statement (EIS), call: 1 (877) 450-6904

For additional information on this Final EIS, contact:

Susan M. Lesica, Document Manager
Office of Nuclear Facilities Management (NE-40)
Office of Nuclear Energy, Science, and Technology
U.S. Department of Energy
19901 Germantown Road
Germantown, MD 20874
Attention: SBSNF EIS
Telephone: (301) 903-8755

For general information on the DOE National Environmental Policy Act (NEPA) process, contact:

Carol M. Borgstrom, Director
Office of NEPA Policy and Assistance (EH-42)
Office of Environment, Safety, and Health
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585
Telephone: (202) 586-4600, or leave a message at: 1 (800) 472-2756

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

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

TABLE OF CONTENTS

SUMMARY

Page

LIST OF FIGURES

LIST OF TABLES

ACRONYMS AND ABBREVIATIONS

ANL-W	Argonne National Laboratory-West
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
FR	*Federal Register*
GMODS	Glass Material Oxidation and Dissolution System
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
MEDEC	melt, drain, evaporate, and calcine (ANL-W process)
NEPA	National Environmental Policy Act
NRC	U.S. Nuclear Regulatory Commission
PUREX	Plutonium-Uranium Extraction
RCRA	Resource Conservation and Recovery Act
SBSNF	Sodium-Bonded Spent Nuclear Fuel
SRS	Savannah River Site

SUMMARY

This document summarizes the U.S. Department of Energy's *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) *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 the facilitation of its disposal in a geologic repository. This environmental impact statement (EIS) was prepared in accordance with the National Environmental Policy Act (NEPA) of 1969, 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), about 40 miles west of Idaho Falls, Idaho; the Enrico Fermi Atomic Power Plant (Fermi-1) in Monroe, Michigan; and the Fast Flux Test Facility at the Hanford site in Richland, Washington. These activities generated approximately 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel for which DOE is now responsible.

Sodium-bonded spent nuclear fuel is distinguished from commercial nuclear reactor spent nuclear fuel by the presence of metallic sodium, a highly reactive material; frequently by metallic uranium and plutonium, which also are 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 treat and manage 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. 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 would be more refined. If the proposed repository at Yucca Mountain in Nevada is developed, final acceptance criteria would not be available until about 2005, when the U.S. Nuclear Regulatory Commission (NRC) would issue a construction authorization. Until such time, the preliminary acceptance criteria are intended to be conservative to allow for uncertainties in the

[1] *Civilian Radioactive Waste Management System - Waste Acceptance System Requirements Document, 1999.*

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

In addition, DOE committed to remove all spent nuclear fuel from Idaho by 2035 in a 1995 agreement with the State of Idaho (Settlement Agreement and Consent Order 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.

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

To ensure that the terms of the State of Idaho Settlement Agreement and Consent Order are met, and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. Appropriate treatment and management of the sodium-bonded spent nuclear fuel would significantly reduce complications related to disposal qualification. Technologies for spent nuclear fuel treatment that could facilitate such qualification therefore should be considered in reaching a decision

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

for treatment of DOE-owned sodium-bonded spent nuclear fuel. Several treatment technologies are at various stages of development and could be used to remove and stabilize the metallic sodium and immobilize or isolate the transuranic and fission products that are in the sodium-bonded spent nuclear fuel. Such technologies include the electrometallurgical treatment process; the PUREX process; placement of the spent nuclear fuel in high-integrity cans; a melt and dilute process; the GMODS process; a direct plasma arc-vitreous ceramic process; and a chloride volatility process.

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

Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section S.1.2.6.3) and in considering the future of PUREX processing capabilities, DOE now needs to decide whether these technologies are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying this NEPA process could result in a loss of capability and of technical staff knowledgeable about and experienced with the demonstration project. This was an important consideration in the decision to proceed with this EIS.

Geologic Repository and Waste Acceptance Criteria

Geologic repositories are deep, excavated underground vaults constructed for the purpose of permanently containing nuclear waste. 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 state 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. Under the schedule for the proposed geologic repository of Yucca Mountain in Nevada, final acceptance criteria would not be available until about 2005, when the NRC would issue a construction authorization. To ensure that the treatment option DOE selects will result in a product that is likely to meet the acceptance criteria, DOE is working with the NRC to obtain comments on the research and development work that DOE will perform to establish treatment technology specifications.

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.

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. NEPA 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 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.

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 NEPA Process

Following the completion of the scoping process, an agency issues a draft EIS for public review and comment. The public comment period on the draft EIS must be at least 45 days in duration, and under DOE's NEPA implementing procedures, at least one public meeting on the draft EIS must be held. DOE must consider all substantive comments on the draft EIS and address these comments in a final EIS. No sooner than 30 days after the notice of availability of the final EIS is issued by the EPA in the *Federal Register,* the agency may issue a Record of Decision; DOE publishes Records of Decision in the *Federal Register.*

This is the final EIS. The draft EIS was issued for public review and comment on July 31, 1999. The public comment period ended on September 28, 1999. In the final EIS, DOE responds to the public comments and, where appropriate, has made revisions to the document based on the public comments.

S.1.2.1 Issues Identified During the Scoping Period

On February 22, 1999, DOE published in the *Federal Register* a Notice of Intent to prepare an *Environmental Impact Statement for Electrometallurgical Treatment of Sodium-Bonded Spent Nuclear Fuel in the Fuel Conditioning Facility at Argonne National Laboratory-West* (64 FR 8553). In this Notice of Intent, DOE invited the public to participate and comment on the 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, electronic mail, a toll-free telephone number, and facsimile. The public scoping comment period began with the publication of the Notice of Intent in the *Federal Register* on February 22, 1999 (64 FR 8553), and ended 45 days later on April 8, 1999.

A total of 228 comments were received during the public scoping comment period. All comments were reviewed and considered by DOE in developing the scope of this EIS. A summary of scoping comments and their disposition is provided in Section A.1 of Appendix A of this EIS. The significant issues identified during the public scoping period are addressed below.

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

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

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

Several commentors asked that the EIS include information about the costs of the proposed action and all of the technology alternatives under consideration. Other commentors stated that the public should have an

NEPA Definitions

NEPA of 1969, as amended: 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 also is considered.

EIS: 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).

opportunity to comment on the independent Nonproliferation Impacts Assessment in the same time frame as the draft EIS, or that this EIS should be delayed until the Nonproliferation Impacts Assessment becomes publicly available. Some suggested that the Nonproliferation Impacts Assessment be included in the EIS. A few commentors expressed the opinion that electrometallurgical treatment of spent nuclear fuel is a proliferation-prone technology.

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

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

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

Comments received during the scoping period were systematically reviewed and evaluated to determine whether the issues raised fell within the scope of the EIS. 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 sodium-bonded spent nuclear fuel types. Thus, several alternatives were 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 conducted a Cost Study and a Nonproliferation Impacts Assessment for the reasonable alternatives. These reports were made available to the public during the public review process.

With respect to comments related to the ongoing Electrometallurgical Treatment Research and Demonstration Project, data from the project were used for the preparation of both the draft and final EIS. The National Research Council issued a final report in April 2000 on the waste forms generated by the technology demonstration. DOE will consider the Council's final report in the Record of Decision process which follows the issuance of the final EIS.

S.1.2.2 Issues Raised During the Public Comment Period on the Draft EIS

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

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

Purpose, Need for, and Timing of the Proposed Action

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

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

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

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

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

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

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

With respect to uranium and plutonium disposition, the EIS states that any uranium that would be separated under the electrometallurgical process would be blended down and stored on site if it originates from driver

spent nuclear fuel, or would be stored on site as depleted uranium if it originates from blanket spent nuclear fuel. The final disposition of the stored uranium has not been decided and is not discussed in this EIS. The disposition of the uranium will be subject to a separate NEPA review. The nuclear nonproliferation policy aspects of this separation are subject to the nuclear nonproliferation policy assessment of the alternatives. The approximately 260 kilograms (572 pounds) of plutonium that would be separated under the PUREX process would be disposed of in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE/EIS-0283) issued in November 1999. This separation is the subject of the Nonproliferation Impacts Assessment, which is independent of this EIS.

Public Availability of Information and Related Documentation

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

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

Cost Issues

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

Comments concerning the costs of the proposed action were considered beyond the scope of the EIS. The EIS was prepared in accordance with NEPA, as well as the Council on Environmental Quality's regulations on implementing NEPA (40 CFR 1500 through 1508) and DOE's NEPA regulations (10 CFR 1021). None of these regulations require the inclusion of a cost analysis in an EIS. The basic objective of the SBSNF EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for treating and managing sodium-bonded spent nuclear fuel and information about their potential impacts on public health and safety and the environment. While cost could be an important factor in the ultimate Record of Decision, the purpose of this EIS is to address the environmental consequences of all alternatives under the proposed action and the No Action Alternative. DOE distributed cost information through the

independent Cost Study released in August 1999, and this information is available to the public on request and in the DOE's public reading rooms. Responses to specific comments related to cost issues are included in Volume 2, Section A.2 of Appendix A of the EIS.

Nuclear Nonproliferation Policy Issues

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

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

Technologies, Alternatives

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

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

Affected Environment and Consequences

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

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

S.1.2.3 Changes from the Draft EIS

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

Results of the Electrometallurgical Treatment Research and Demonstration Project

As a result of public concern that results of the demonstration project were not incorporated in the draft EIS, a section (Section 1.6.3) was added in the final EIS with a description, status, and results of the demonstration project (see also Section S.1.2.6.3).

Justification of Purpose and Need and Timing

As a result of public concern that the draft EIS did not adequately justify the need and timing for the proposed action, Section 1.2 of the final EIS was revised to reflect DOE's position and DOE's responses to the related comments (see also Section S.1.1).

Relationship to Other NEPA Actions

Section 1.6.2.2 of the final EIS was revised to update the information provided on the *Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada,* which was issued in July 1999 (see also Section S.1.2.6.2). A discussion of the Surplus Plutonium Disposition Final EIS and Record of Decision was added to Section 1.6.1.8 of the final EIS (see also Section S.1.2.6.1).

Sodium Removal and Disposition

As a result of public comment, the description of an alternate method for decladding and cleaning sodium-bonded blanket spent nuclear fuel, the laser declad and alcohol wash process, was added in Section 2.3.9 of the final EIS (see also Section S.3.9). The reason the process was not included in the evaluation of the reasonable alternatives also is in these sections.

No Action Alternative Definition

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

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

No Action Alternative Assumptions

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

Dose and Risk Calculations

As a result of public comments and the availability of recent data from the Electrometallurgical Treatment Research and Demonstration Project and the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement,* dose calculations were revised in the final EIS. These revisions include: (1) the addition of project total doses to workers, (2) the project total risk to the public under normal operations, and (3) changes in doses and risks to the public and workers from accidents. In addition, dose and risk values were rounded, resulting in some changes in the numerical values in the EIS.

Air Quality

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

Land Use/Ecology

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

Water Quality

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

Geology and Soils

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

Existing Human Health Risk

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

Waste Management

Records of Decision for the *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* addressing the management of high-level radioactive waste and low-level radioactive waste were issued on August 26, 1999 (64 FR 46661), and February 25, 2000 (65 FR 10061), respectively. A summary of these decisions was added to the waste management discussion for both INEEL and SRS (Sections 3.2.11 and 3.3.11, respectively, of the final EIS).

Cumulative Impacts

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

Electric Energy Consumption

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

Settlement Agreement and Consent Order with the State of Idaho

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

Melt and Dilute Driver Fuel at SRS

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

Preferred Alternative

In accordance with requirements of the Council on Environmental Quality regulations (40 CFR 1502.14e), the final EIS incorporates DOE's Preferred Alternative for the treatment and management of sodium-bonded spent nuclear fuel. The Preferred Alternative is discussed in Section 2.8 (see also Section S.5.8).

Transportation

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

Miscellaneous Revisions and Editorial Changes

Several sections in the SBSNF Final EIS were revised to reflect the availability of more recent data or to include corrections, improvements in the presentation, and other editorial changes. Among these are a simplified presentation of the consequences (Section S.8) and the addition of cumulative impacts (Section S.9) in the Summary. None of these revisions affects the environmental impact analysis presented in the EIS.

S.1.2.4 Scope of this EIS

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

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

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

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

S.1.2.5 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 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 Impacts Assessment, the 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. DOE could make a different decision for each type of sodium-bonded spent nuclear fuel.

S.1.2.6 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.1.2.6.1, ongoing actions are described in Section S.1.2.6.2, and the Electrometallurgical Treatment Research and Demonstration Project in Section S.1.2.6.3.

S.1.2.6.1 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 the Idaho National Engineering Laboratory (now known as INEEL). For programmatic spent nuclear fuel management, this document analyzed alternatives that included no action, decentralization, regionalization, centralization, and the use of plans that existed in 1992 and 1993 for the management of these materials. For INEEL, this document analyzed alternatives such as no action, a 10-year plan, and minimum and maximum treatment, storage, and disposal of DOE waste. The SBSNF EIS was 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) waste generated at SRS. This waste would come from ongoing operations and potential actions, new missions, environmental restoration, and decontamination and decommissioning programs. The SRS Waste Management EIS 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 spent nuclear fuel assemblies and up to 25 EBR-II blanket spent nuclear fuel assemblies in the Fuel Conditioning Facility at ANL-W. As noted in the environmental assessment, DOE had identified electrometallurgical treatment as a promising technology to treat EBR-II spent nuclear fuel, but an appropriate demonstration was needed to provide DOE with sufficient information to evaluate its technical feasibility. The successful demonstration of the electrometallurgical treatment technology on EBR-II spent nuclear fuel, combined with research and testing of the resulting waste forms, provides DOE with the information needed to determine whether this treatment

technology should be used to treat the remainder of EBR-II spent nuclear fuel and/or other types of spent nuclear fuel. The demonstration project is discussed in Section S.1.2.6.3.

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 the disposition and 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)

The Waste Management Programmatic EIS examined the potential environmental and cost impacts of strategic management alternatives for managing five types of radioactive and hazardous waste that have resulted and will continue to result from nuclear defense and research activities at a variety of sites around the United States. The five waste types are mixed waste, low-level radioactive waste, transuranic waste, high-level radioactive waste, and hazardous waste. The 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 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)

The Advanced Mixed Waste EIS assessed the potential environmental impacts associated with four alternatives related to the construction and operation of the Advanced Mixed Waste Treatment Facility at INEEL. The Advanced Mixed Waste Treatment Facility will treat transuranic waste, mixed waste, and alpha-contaminated mixed waste, at INEEL 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 based on DOE's Waste Management Programmatic EIS.

Surplus Plutonium Disposition Final Environmental Impact Statement (DOE/EIS-0283, November 1999)

The Surplus Plutonium Disposition EIS evaluated reasonable alternatives for the siting, construction, and operation of facilities required to implement DOE's disposition strategy for surplus plutonium. The facilities analyzed include a pit disassembly and conversion facility, a plutonium conversion and immobilization facility, and a mixed oxide fuel fabrication facility. The Surplus Plutonium Disposition EIS also analyzed the potential impacts of fabricating a limited number of mixed oxide fuel assemblies for testing in a reactor. DOE selected SRS as the location for all of these facilities. The Surplus Plutonium Disposition EIS addresses the disposition of material that the SBSNF EIS could generate.

S.1.2.6.2 Ongoing NEPA Actions

Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement (DOE/EIS-0279, March 2000)

The SRS Spent Nuclear Fuel Management EIS analyzed the potential impacts from the safe and efficient management of spent nuclear fuel and targets assigned to SRS, including placing these materials in forms

suitable for ultimate disposition. Options to treat, package, and store spent nuclear fuel are discussed in this document. The alternatives considered in the SRS Spent Nuclear Fuel Management EIS encompass a range of new packaging, new processing, and conventional processing technologies for the treatment of spent nuclear fuel. Most of these processing technologies are also analyzed in this SBSNF EIS. The preferred alternative identified in the SRS Spent Nuclear Fuel Final Management EIS would prepare nearly all of the spent nuclear fuel at SRS for disposition using a melt and dilute treatment process. The remaining material would be managed using chemical separation. The SRS Spent Nuclear Fuel Management 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.

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-0250D, July 1999)

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

Under the No Action Alternative, this EIS evaluates the potential impacts of the continued storage of spent nuclear fuel and high-level radioactive waste at the current storage locations using two scenarios: the first assumes continued storage under institutional controls for at least 10,000 years, and the second assumes no institutional controls after 100 years. This EIS also evaluates the disposal at the proposed repository of spent nuclear fuel or high-level radioactive waste that may be generated by the proposed action presented in the SBSNF EIS.

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

This draft EIS was issued in December 1999. It evaluates alternatives for managing the high-level radioactive waste and associated radioactive waste and facilities at INEEL. Under the terms of the 1995 Settlement Agreement and Consent Order with the State of Idaho, DOE agreed to treat high-level radioactive waste currently stored at INEEL and to prepare the waste in a form ready to be shipped out of the State of Idaho by 2035. The purpose of this EIS is to assist DOE in making decisions concerning the management of this radioactive waste to ensure compliance with applicable laws and regulations, and protect the environment and the health and safety of the workers and the public in a cost-effective manner. The proposed action under this EIS contributes to the cumulative impacts at the site discussed in the SBSNF EIS.

S.1.2.6.3 Electrometallurgical Treatment Research and Demonstration Project

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

using a fraction of the EBR-II spent nuclear fuel. Following the National Research Council's recommendation, DOE conducted an environmental assessment of the demonstration project. The environmental assessment was completed in May 1996 and resulted in a Finding of No Significant Impact, so that no further NEPA review was necessary for the demonstration project to proceed.

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

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

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

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

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

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

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

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

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

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

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

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 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 made from depleted uranium, a type of uranium in which most of the fissile uranium-235 has been removed. Blanket fuel is usually placed at the outer perimeter of the core and is used to breed plutonium-239, a fissile material. Blanket fuel contains primarily 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**.

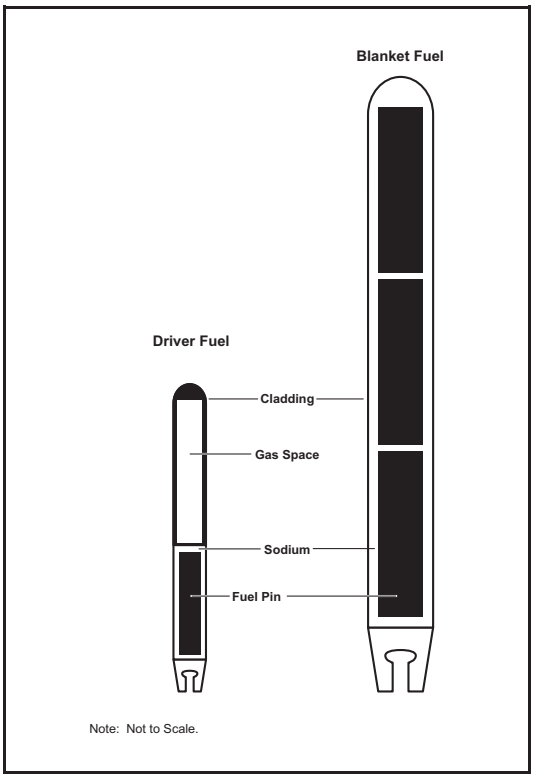

Figure S–1 Typical Driver and Blanket Spent Nuclear Fuel Elements

The blanket and driver spent nuclear fuel 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 the fuel and cladding. This allows mechanical stripping of the blanket spent nuclear fuel cladding.

S.2.1 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). 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 for some fuel) fuel elements are put together in a stainless steel hexagonal duct to make a fuel assembly approximately 2.3 meters (92 inches) long and 5.8 centimeters (2.3 inches) across.

Sodium-Bonded Spent Nuclear Fuel

Driver Fuel: During irradiation, the fuel expands. Metallic sodium enters the fuel and becomes mechanically inseparable from the uranium. In addition, the fuel and cladding mix so that mechanical removal of the cladding is not practical.

Blanket Fuel: When irradiated, only minimal amounts of sodium enter the fuel, and there is little or no mixing of the fuel and the cladding. Blanket fuel could be declad and the sodium removed.

Because of these differences between irradiated driver fuel and blanket fuel, the alternatives propose different treatment methods for each fuel type.

The EBR-II blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in metallic form. In EBR-II, the blanket spent nuclear fuel 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 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. Because of its lower burnup, the Fermi-1 blanket fuel, which contains only about 0.2 percent plutonium by weight compared to approximately 1 percent plutonium by weight for the EBR-II blanket fuel, is subject to less stringent safeguard and security requirements than the EBR-II blanket fuel. This is an important consideration in the cost of managing the storage of these two fuel types.

After the Fermi-1 reactor was permanently shut down, the blanket assemblies were placed into 14 canisters and transported to the Idaho Nuclear Technology and Engineering Center (INTEC)—formerly known as the Idaho Chemical Processing Plant (CPP)—in 1974 and 1975 in 14 shipments. The canisters, made of stainless steel with a carbon steel basket inside, were placed into CPP-749, an underground dry storage system. 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 Miscellaneous 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 in 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 fuel consists of uranium and/or plutonium carbides, nitrides, and oxides in addition

Sources of Sodium-Bonded Spent Nuclear Fuel

EBR-II is a 62.5-megawatts-thermal research and test reactor located at ANL-W that was 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 33 years of operation, numerous fuel designs were tested in EBR-II, and EBR-II spent nuclear fuel contains both driver and blanket fuel.

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.

The **Fast Flux Test Facility Reactor**, located on the Hanford site near Richland, Washington, is a 400-megawatts-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 fuel from the Fast Flux Test Facility Reactor is experimental fuel.

to metal uranium or alloy. For the purposes of this EIS, this miscellaneous fuel is assumed to have driver fuel characteristics. This fuel is stored at several DOE sites such as the Hanford Site, Oak Ridge National Laboratory, 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

Spent Nuclear Fuel Type	*Storage Volume* [a] *(cubic meters)*	*Metric Tons of Heavy Metal*	*Sodium Content (kilograms)*
EBR-II driver	58 [b]	3	83
EBR-II blanket	13	22	176
Fermi-1 blanket	19	34	365
Fast Flux Test Facility driver	8 [b]	0.3	7
Miscellaneous [c]	3 [b]	0.1	31
Total	101	60	662

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

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

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

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 disposal in a geologic repository. These are: the electrometallurgical treatment; the PUREX process; packaging in high-integrity cans; the melt and dilute process; the GMODS process; the direct plasma arc-vitreous ceramic process; and the chloride volatility process. Each of these methods is discussed below. In formulating the reasonable alternatives under the proposed action, the GMODS process, the direct plasma arc process, and the chloride volatility process were not considered sufficiently mature technologies to be included as reasonable alternatives (see Section S.6).

Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository without treatment, i.e., packaging the fuel in high-integrity cans without sodium removal, has been considered in this EIS under the No Action Alternative.

S.3.1 Electrometallurgical Treatment Process

The electrometallurgical treatment process was developed at 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, nitride, 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 a high temperature 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. 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 disposal in a geologic repository.

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 metallic high-level radioactive waste form that is expected to be suitable for disposal in a geologic repository.

S.3.2 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 oxide, plutonium metal, and high-level radioactive waste. The uranium oxide would be stored on site as depleted uranium. The plutonium would be disposed of in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE/EIS-0283).

S.3.3 High-Integrity Can Packaging

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 corrosion protection during storage, prior to repository transfer. It also could provide long-term protection in a repository (i.e., for 1,000 or more years after repository closure with no institutional control). The high-integrity cans are placed into standardized canisters that are ready for disposal in waste packages. High-integrity cans would be used to store the sodium-bonded spent nuclear fuel on site until it can be shipped to a repository.

The analysis for packaging sodium-bonded spent nuclear fuel in high-integrity cans was performed with and without decladding and/or sodium removal. Packaging sodium-bonded blanket spent nuclear fuel in high-integrity cans with sodium removal was analyzed in the EIS under Alternative 2. Packaging sodium-bonded spent nuclear fuel in high-integrity cans without sodium removal was considered in this EIS as a direct disposal option under the No Action Alternative (see Sections S.3.8 and S.5.1). The high-integrity cans would be placed in dry storage at ANL-W. They would be placed into a standardized canister for transportation and eventual placement in waste packages in a geologic repository.

S.3.4 Melt and Dilute Process

The melt and dilute process involves chopping and melting the spent nuclear fuel and diluting it 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 the 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 the sodium and 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 (132 pounds) of material. The canisters would be stored until they fit into the processing schedule. Following validation of the contents, the canisters would be loaded into a melting furnace with additional aluminum, 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. Radioactive waste steel would be added. This furnace would be 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 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 spent nuclear metallic fuel. Uranium nitride, oxide, and carbide fuel types cannot be treated with this process because of their high melting points.

S.3.5 GMODS Process

The GMODS process uses oxides to convert unprocessed spent nuclear fuel directly to borosilicate glass. The basic concept is to combine unprocessed sodium-bonded spent nuclear fuel and lead-borate glass in a glass melter at a 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, if a research and development demonstration project shows that the process can 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, if a research and development demonstration project shows that the process can deal with sodium and other factors.

Metallic fuel such as the 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

For the purpose of this EIS, direct disposal of sodium-bonded spent nuclear fuel is disposal without sodium removal. The sodium-bonded spent nuclear fuel (driver and blanket) would be packaged in high-integrity 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 preclosure operations. However, one of the key requirements, as specified in the current April 1999 version of DOE's Waste Acceptance Systems Requirements Document and in NRC requirements for acceptance of spent nuclear fuel or high-level radioactive waste in a geologic repository, is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective [10 CFR 60.135(b)(1)]. Under direct disposal, the pyrophoric and chemically reactive metallic sodium would not be removed and, therefore, the direct disposal option would not meet current DOE or NRC repository acceptance criteria.

S.3.9 Sodium Removal and Disposition

As discussed in Section S.2 and the preceding sections, the disposition of the metallic sodium in the sodium-bonded spent nuclear fuel varies with the treatment method. For those methods that do not require the removal of metallic sodium prior to treatment, or decladding of the fuel (e.g., the electrometallurgical process), the metallic sodium would be converted into a nonreactive salt as part of the process and would be incorporated in the high-level radioactive waste product of the process.

For the methods that require the removal of sodium prior to treatment and/or decladding of the 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. Decladding and sodium removal could be done using either a mechanical process (the melt, drain, evaporate, and calcine [MEDEC] process) or a laser declad and alcohol wash process.

In the MEDEC process, blanket fuel is brought into an argon-atmosphere hot cell where the ends of the cladding are cut off. The fuel is subjected to a temperature of about 200 °C (390 °F), causing melting of the sodium, which is drained into a collection tank. The temperature is raised to about 500 °C (930 °F), which volatilizes the residual sodium. This sodium exits the fuel as sodium vapor, which is condensed in a trap and collected.

To remove the cladding after the 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 previously were cut off. Experience with unirradiated blanket spent nuclear fuel at Argonne National Laboratory has shown that the pins could be mechanically pushed out of the stainless steel cladding after all the sodium has been eliminated.

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

In the laser declad and alcohol wash process, which has been demonstrated at Rockwell International Hot Laboratory in California, a remote operation laser system and cutting machine would be used to cut the fuel elements in a predefined sequence within a hot cell. The fumes generated during the cutting process would be filtered and exhausted through an off-gas system. The fuel pins along with the cladding strips would be washed in an alcohol/water mixture to neutralize the metallic sodium and fission product (i.e., cesium) contamination. The alcohol/water solution would be partially evaporated, and the sodium/cesium alcoholates and hydroxides would be neutralized, then solidified in a grouting agent, and disposed of as low- or high-level radioactive waste, depending on the cesium content.

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

Table S–3 summarizes sodium removal and 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 radioactive ceramic waste product of the process.
High-integrity cans Blanket fuel	No	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
PUREX process Blanket fuel	Yes	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
Melt and dilute process Driver fuel	No	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste. The remaining sodium is separated during the process, converted to nonreactive ceramic waste form, and disposed of as high-level radioactive waste.
Blanket fuel	Yes [a]/No [b]	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
Direct disposal [c] Blanket and driver fuel	No	No	Disposed of in metallic reactive form in high-integrity cans.

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

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 ANL-W

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

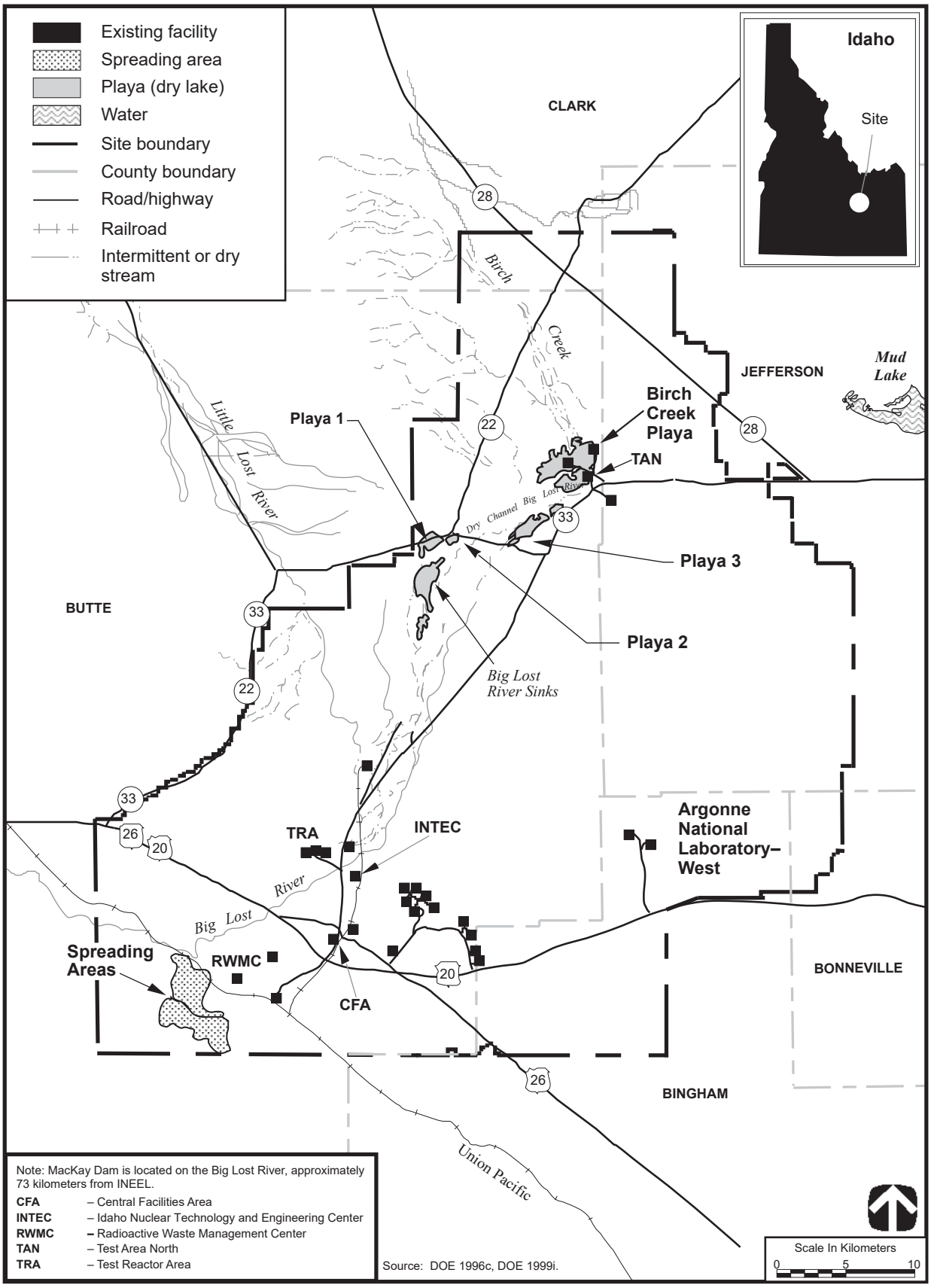

Figure S–2 Map of INEEL

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 also is 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 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.

ANL-W

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 currently is 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 the main ANL-W facilities within the ANL-W boundary, the Radioactive Scrap and Waste Facility looks somewhat like a large parking lot on the surface. The facility has a permit issued by the State of Idaho for interim storage of mixed waste regulated under the Resource

Conservation and Recovery Act (RCRA). The Radioactive Scrap and Waste Facility provides protection against corrosion for the more than 1,000 underground steel liners available for waste storage of materials handled at ANL-W.

INTEC is located northeast of the Central Facilities Area at INEEL. It is one of the sites where the sodium-bonded spent nuclear fuel currently is 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 radioactive waste, environmental restoration and decontamination and dismantling activities, and technology development. For the proposed action, the facility would be used to continue storing sodium-bonded spent nuclear fuel and for packaging the treated or untreated sodium-bonded spent nuclear fuel in standardized canisters in preparation for transport and disposal in a geologic repository. However, because it has no hot cell with an inert gas atmosphere, INTEC cannot be used for any sodium removal activities under the proposed action.

S.4.2 SRS

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 separation 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 shut down permanently. 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 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. 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.

The F-Canyon Complex at SRS

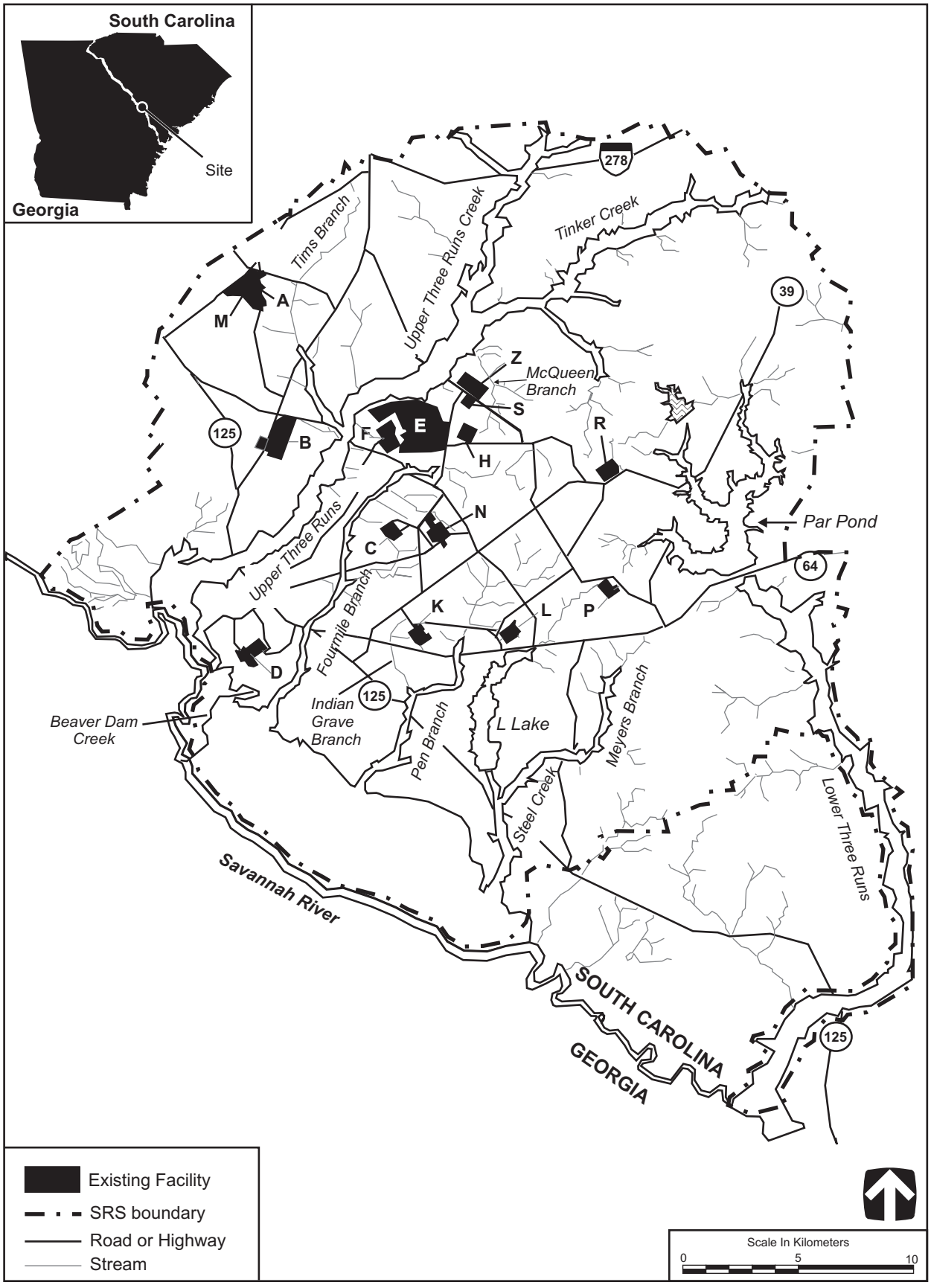

Figure S–3 Map of SRS

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 shut-down 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 in implementing this technology would be the design of an off-gas system to capture volatilized fission products. Preliminary engineering studies indicate that the system could be designed using proven approaches for managing off-gases. The impacts from the construction of a melt and dilute facility at SRS's Building 105-L are addressed in the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement*.

L-Reactor Complex at SRS

The Defense Waste Processing Facility, located in the S-Area, 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 No Action Alternative, the alternatives under the proposed action, and the Preferred Alternative are discussed below. Although each alternative addresses both blanket and driver spent nuclear fuel, DOE considered the blanket fuel and driver fuel separately in identifying the Preferred Alternative and will do the same for the subsequent Record of Decision. In other words, DOE is considering all combinations of technologies, options, and fuel types, not only the specific combinations that are explicitly discussed in the EIS.

S.5.1 No Action Alternative

Under the No Action Alternative, all or some part of the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed), except for stabilization activities that may be necessary to prevent

Proposed Action—DOE proposes to treat and manage sodium-bonded spent nuclear fuel and facilitate its ultimate disposal in a geologic repository.

Alternative 1. *Driver and Blanket Fuel:* Under this alternative, both driver and blanket fuel would be treated using electrometallurgical treatment at ANL-W.

Alternative 2. *Driver Fuel:* Electrometallurgical treatment would be used to treat the driver fuel at ANL-W. *Blanket Fuel:* The sodium from the blanket fuel would be removed without decladding and the elements would be packaged in high-integrity cans. Sodium removal and packaging would occur at ANL-W.

Alternative 3. *Driver Fuel:* Electrometallurgical treatment would be used to treat the driver fuel at ANL-W. *Blanket Fuel:* The fuel pins would be separated from the cladding and cleaned to remove metallic sodium at ANL-W. The cleaned fuel pins would be shipped to SRS for treatment using the PUREX process at the F-Canyon facility.

Alternative 4. *Driver Fuel:* Electrometallurgical treatment would be used to treat the driver fuel at ANL-W. *Blanket Fuel:* The metallic sodium would be removed without decladding. Then the elements would be treated using the melt and dilute process. All treatment would occur at ANL-W.

Alternative 5. *Driver Fuel:* Electrometallurgical treatment would be used to treat the driver fuel at ANL-W. *Blanket Fuel:* The fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W. Then they would be shipped to Building 105-L at SRS and treated using the melt and dilute process.

Alternative 6. *Driver and Blanket Fuel:* Under this alternative, both the driver and blanket fuel would be treated at ANL-W using the melt and dilute process. The melt and dilute process would be modified slightly for each fuel type.

No Action Alternative —Under the No Action Alternative, two options were analyzed:

- Continued storage until 2035 or until development of a new or currently less mature technology to treat all or part of the sodium-bonded spent nuclear fuel
- Direct disposal of the sodium-bonded spent nuclear fuel in high-integrity cans

potential degradation of some of the spent nuclear fuel. Under the No Action Alternative, two options were analyzed: (1) the sodium-bonded spent nuclear fuel would continue to be stored until 2035 at its current location, subject only to activities dictated by the amended Record of Decision (61 FR 9441) for the Programmatic Spent Nuclear Fuel EIS and other existing site-specific NEPA documentation, or until another technology, currently dismissed as a reasonable alternative because it is less mature (e.g., GMODS or plasma arc) is developed; and (2) the sodium-bonded spent nuclear fuel would be disposed of directly in a geologic repository without treatment. The fuel would be packaged in high-integrity cans without sodium removal. Under the latter option, the sodium-bonded spent nuclear fuel at INTEC would be transported to ANL-W for packaging. As discussed in Section S.3.8, direct disposal would not meet current DOE or NRC repository acceptance criteria requirements.

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

could lead to significant impacts to the environment and the health and safety of the public from radioactive releases caused by the gradual degradation of the fuel and its containment.

In selecting the No Action Alternative, DOE could actively pursue research and development of another treatment technology including, for example, the GMODS and plasma arc methods. These methods offer the potential to treat both blanket and driver 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.6.

S.5.2 Alternative 1: Electrometallurgically Treat 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 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 spent nuclear fuel would be separated from the assembly hardware and packaged and transferred to the Fuel Conditioning Facility for electrometallurgical treatment. The separated hardware would be packaged and managed as low-level radioactive waste.

After treatment, the low-enriched uranium by-product 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 afterward 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 7 years.

S.5.3 Alternative 2: Clean and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the blanket spent nuclear fuel elements (approximately 57 metric tons of heavy metal) would be packaged in high-integrity stainless steel cans at ANL-W after removal of the sodium without decladding, as discussed in Section S.3.9. Removal of the sodium from the sodium-bonded blanket spent nuclear fuel would take place at the Hot Fuel Examination Facility at ANL-W. The packaging in high-integrity cans would take place in the same facility. The high-integrity cans would be transferred to the Radioactive Scrap and Waste Facility for storage.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated using the electrometallurgical treatment process described in Section S.5.2 (Alternative 1).

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. In addition, some new waste

handling equipment would be needed for the electrometallurgical treatment of the driver sodium-bonded spent nuclear fuel.

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 could start in 2000 and could be completed in approximately seven years.

S.5.4 Alternative 3: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; PUREX Process Blanket Fuel at SRS

Under this alternative, the blanket spent nuclear fuel pins (approximately 57 metric tons of heavy metal) would be packaged in aluminum cans and shipped to SRS for treatment using the PUREX process at the SRS F-Canyon facility. The blanket spent nuclear fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W, as discussed in Section S.3.9.

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. Approximately 260 kilograms (572 pounds) of separated plutonium would be immobilized using the "can-in-canister" technology at SRS for eventual geologic repository disposal in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Environmental Impact Statement* (DOE/EIS-0283). Considering the commitment of F-Canyon to other DOE missions, PUREX processing of the blanket spent nuclear fuel would start no earlier than 2005 and would last less than one year. Decladding and sodium removal activities at ANL-W would not start earlier than 2003. Therefore, these activities would determine the length of the process.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process described in Section S.5.2 for Alternative 1. As in the case of Alternative 2, electrometallurgical treatment of the driver spent nuclear fuel could start in 2000 and could be completed in approximately seven years.

S.5.5 Alternative 4: Melt and Dilute Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the blanket spent nuclear fuel elements (approximately 57 metric tons of heavy metal) would be treated at ANL-W using the melt and dilute Option 2 process described in Section S.3.4. Prior to treatment, the metallic sodium would be removed without decladding at ANL-W, as discussed in Section S.3.9. Removal of the sodium from the sodium-bonded blanket spent nuclear fuel could take place at the Hot Fuel Examination Facility at ANL-W. Equipment for sodium removal would need to be installed at the facility. Equipment necessary for the melt and dilute process also would need to be installed at the facility, including the addition of a melter and an off-gas system. Metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be transferred to the Radioactive Scrap and Waste Facility for storage pending ultimate disposal. 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 eight years.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process described in Section S.5.2 for Alternative 1. Treatment of the driver spent nuclear fuel could start as early as 2000 and could be completed in approximately seven years.

S.5.6 Alternative 5: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; Melt and Dilute Blanket Fuel at SRS

Under this alternative, the blanket spent nuclear fuel pins (approximately 57 metric tons of heavy metal) would be packaged and shipped to SRS for treatment. The blanket spent nuclear fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W. The declad and cleaned blanket spent nuclear fuel pins would be received at Building 105-L at SRS and treated using the melt and dilute Option 1 process, as described in Section S.3.4. Decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal would take place at the Hot Fuel Examination Facility at ANL-W. After decladding and sodium removal, the blanket spent nuclear fuel pins would be packaged and stored temporarily at the Hot Fuel Examination Facility pending shipment to SRS.

At SRS, the cans containing the blanket spent nuclear fuel pins would be unpacked at Building 105-L and the blanket spent nuclear fuel pins would be treated using the melt and dilute process. For the purpose of evaluating this alternative, it is assumed that the melt and dilute facility is operational at SRS, as proposed in the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement*. Metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be stored at the L-Area storage pending ultimate disposal.

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.

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 for Alternative 1. Treatment of the driver spent nuclear fuel at ANL-W could start in 2000 and could be completed in approximately seven years.

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. Uranium nitride, oxide, and carbide fuel types cannot be treated using the melt and dilute process because of their high melting points.

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

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

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

S.5.8 Preferred Alternative

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

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

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

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

For several years, DOE has been actively developing electrometallurgical treatment technology specifically for the management of sodium-bonded spent nuclear fuel. Having completed a successful demonstration of electrometallurgical treatment, DOE believes that this technology has the highest probability of meeting the Department's needs for managing much of the sodium-bonded spent nuclear fuel. Electrometallurgical technology would convert the reactive fuel into ceramic and metallic waste forms, both of which are more stable than untreated sodium-bonded spent nuclear fuel. In addition, uranium would be separated from the spent nuclear fuel, blended with depleted uranium if needed to lower enrichment levels, and cast into ingots to be stored until a disposition decision is made through a separate NEPA review. Most of the plutonium would be disposed of in the ceramic waste form, with the remaining small fraction disposed of in the metallic waste form. Currently, the only waste form that has been tested and analyzed extensively under geologic repository conditions and may be accepted for repository disposal is borosilicate glass. Tests have shown

the ceramic and metallic waste forms from electrometallurgical treatment may perform as well as the standard borosilicate glass waste form. The ceramic and metallic waste forms would require less storage volume than untreated spent nuclear fuel.

S.6 Alternatives Considered and Dismissed

In identifying the reasonable alternatives for evaluation in this EIS, two separate issues led to the determination of alternatives that were considered and dismissed: (1) the level of maturity of the alternative technologies, and (2) the level of effort required to modify an existing facility to implement a specific technology. The construction of new facilities when existing facilities are still operative was considered not 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 proven successfully 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. 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 fuel. In addition, unlike the other technologies that would not require new construction, both of these technologies (i.e., GMODS and plasma arc) would require the installation of large, specialized equipment in new hot cell facilities, the size and complexity of which are not determined sufficiently to allow detailed environmental impact analysis.

GMODS Process

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

GMODS would require large, specialized equipment to be installed in eight new large hot cell facilities. GMODS would dissolve the fuel elements or fuel assemblies entirely in a lead/lead-oxide system. An off-gas treatment system similar to that for the melt and dilute process would be required to treat the radioactive elements volatilized at about 1,000 °C (1,830 °F). The GMODS equipment could produce an intermediate waste form containing most of the actinides, fission products, and structural materials. After some preprocessing, the waste stream would be fed into the melter for the production of a new type of borosilicate glass log. These logs would contain uranium, other actinides, and structural elements, in addition to the fission products.

Because of the highly corrosive nature of the chemicals in the system, the technical feasibility of the alternative has not been established. This would add an additional degree of uncertainty to the waste estimates, as well as to the ultimate success of the fuel conditioning project.

Direct Plasma Arc-Vitreous Ceramic Process

The plasma arc-vitreous ceramic process is being used for the vitrification of low-level mixed waste. However, vitrification of spent nuclear fuel by this process is understood only on a conceptual level. The plasma arc treatment method would require large, complex equipment to be installed in a new, specially

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

The high temperature 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 Test Area North would require extensive modification to treat sodium-bonded spent nuclear fuel. Implementation of this alternative would require the construction of an argon hot cell. In addition, it would require either the procurement of new equipment or the transfer of already-contaminated equipment and other systems existing at ANL-W.

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

As discussed in Section 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 of Driver Spent Nuclear Fuel Using SRS Melt and Dilute Process

As discussed in Section S.3.4, the treatment of driver spent nuclear fuel would require a modified melt and dilute process that would be capable of handling the sodium volatilized from processing chopped driver spent nuclear fuel elements with the cladding intact. To accomplish this at SRS, significant design changes would be required from the process that DOE has proposed for the aluminum-clad spent nuclear fuel, which does not contain sodium. These design changes do not appear justified for the sole purpose of treating the relatively small quantity of driver spent nuclear fuel.

Treatment Using INEEL PUREX Process

Sodium-bonded spent nuclear fuel from EBR-II was being processed at the Idaho CPP (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.7 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.8 CONSEQUENCES

This section summarizes the environmental impacts associated with the No Action Alternative and the six alternatives under the proposed action that are evaluated in detail in this EIS. For the No Action Alternative and the six alternatives evaluated, the proposed facilities already exist. Except for internal building modifications and new equipment installation, no construction activities would be required. Therefore, DOE has determined the proposed action would have minimal or no impacts on land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources. These areas were not evaluated in detail in the EIS. The information presented below is based on Chapter 4 of the EIS, 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. A summary of the environmental impacts for the No Action Alternative and the six alternatives under the proposed action is presented as **Table S–4**.

As explained below and shown in Table S–4, for the alternatives evaluated the analyses showed that there would be no significant impacts on air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, and transportation. The radiological and nonradiological gas and liquid releases, as well as the associated exposures to workers and the public, would be well below regulatory standards and guidelines and, therefore, no mitigation measures would be warranted. Finally, the environmental impact analysis indicates that there are no significant impacts that would discriminate one alternative over another from an environmental impact point of view.

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 (i.e., induced cancers) 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. Because of the delayed effect, the cancers are referred to as "latent" cancers.

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 incidents, are presented in this EIS.

The number of latent cancer fatalities is estimated using risk factors determined by the International Commission on Radiological Protection. A risk factor is the probability that an individual would incur a latent cancer fatality during his or her lifetime if the individual receives a unit of radiation dose (1 rem). The risk factor for workers is 0.0004 (latent cancer fatality per rem), and 0.0005 (latent cancer fatality per rem) for individuals among the general public. The risk factor for the public is slightly higher because the public includes infants and children, who are more sensitive to radiation than adults.

Examples:

> The latent cancer fatality risk for an individual (nonworker) receiving a dose of 0.1 rem would be 0.00005 (0.1 rem x 0.0005 latent cancer fatality per rem). This risk can also be expressed as "0.005 percent chance" or "one chance in 20,000."
>
> The same concept is used to calculate the latent cancer fatality risk from exposing a group of individuals to radiation. The latent cancer fatality risk for individuals in a group of 100,000, each receiving a dose of 0.1 rem, would be 0.00005, as indicated above. This individual risk, multiplied by the number of individuals in the group, expresses the number of latent cancer fatalities that could occur among the individuals in the group. In this example, the number would be 5 latent cancer fatalities (100,000 x 0.00005). A number of latent cancer fatalities less than 1 means that the radiation exposure is not sufficient to cause a single latent cancer fatality among the members of the group. In this case, the risk is expressed as a probability that a single latent cancer fatality would occur among the members of the group. For example, 0.05 latent cancer fatalities can be stated as " there is one chance in 20 (1/0.05) that one latent cancer fatality would occur among the members of the group."

The EIS provides estimates of probability of a latent cancer fatality occurring for the involved and noninvolved workers, the maximally exposed offsite individual, an average individual, and the general population. These categories are defined as follows:

Involved worker—An individual worker participating in the operation of the facilities.

Noninvolved worker—An individual worker at the site other than the involved worker.

Maximally exposed offsite individual—A member of the public residing at the site boundary who could receive the maximum dose of radiation or exposure to hazardous chemicals.

Average individual—A member of the public receiving an average dose of radiation or exposure to hazardous chemicals.

Population—Members of the public residing within an 80-kilometer (50-mile) radius of the facility.

Air Quality

The proposed action and either option of the No Action Alternative would have a negligible impact on existing air quality at ANL-W, INTEC, or SRS. Radiological emissions also would be low and well below regulatory concern (see the public and occupational health and safety discussion, below).

Water Resources

Surface water is not used at ANL-W or INTEC, and this would not change under the proposed action or either option of the No Action Alternative. Groundwater use, primarily domestic consumption, would remain at current levels under the proposed action and could decrease with a reduction in workers at ANL-W under the No Action Alternative.

No changes are expected in liquid effluent discharges under the proposed action or the No Action Alternative at ANL-W. There currently are no discharges to surface waters (radiological or nonradiological), except for discharges of nonhazardous liquid waste, which are monitored and subject to Idaho Land Permit Application 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, cesium, promethium, 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

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

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 risk to the health and safety of the workers and the public under the proposed action or 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. As indicated below and in Table S-4, this risk would be very small. The health effects presented below are given in terms of total project effects to account for the variations in annual exposures within an alternative as well as variations in project durations between alternatives.

Radiological Exposures

The maximum project total impact to the population within 80 kilometers (50 miles) from radioactive releases under the proposed action would be 0.000012 (Alternative 6) additional latent cancer fatalities, or one chance in 83,000 for one additional latent cancer fatality within the exposed population, compared to 6.5×10^{-6} latent cancer fatalities for the No Action Alternative. The project total dose corresponding to this

impact would be 0.024 person-rem. For comparison purposes, the collective dose to this population from natural background radiation for a single year would be 86,250 person-rem.

The maximum project total impact to the maximally exposed offsite individual under the proposed action would be 2.0×10^{-9} (Alternative 6), or one chance in 500 million that this individual would develop a fatal cancer, compared to 1.1×10^{-9} (one chance in 900 million) for the No Action Alternative. The maximum project total dose to this individual under the proposed action would be 0.004 millirem. The regulatory limit for offsite individuals is 10 millirem per year from air exposure and 100 millirem per year from all pathways.

The maximum project total impact to the workers under the proposed action at ANL-W or SRS would be 0.13 additional latent cancer fatalities, or one chance in 7 that the work force would experience an additional latent cancer fatality from this exposure, compared to 0.084 (one chance in 12) additional latent cancer fatalities expected from the No Action Alternative. The maximum project total dose under the proposed action would be 319 person-rem for the worker population. DOE's administrative limit for individual worker exposure is 2,000 millirem per year.

Hazardous Chemical Exposures

Hazardous chemical impacts resulting from the proposed action or either option under the No Action Alternative under normal conditions would be small because any routine emissions of hazardous chemicals would be very low.

Hazardous chemical impacts under accident conditions, evaluated in terms of Emergency Response Planning Guideline values, indicate that under the worst postulated accident conditions, no adverse health effects to a worker or the maximally exposed offsite individual at either ANL-W or SRS would occur.

Environmental Justice

As discussed above, the impacts from the proposed action or 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 the region.

Waste Management

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

Table S–4 also presents a comparison of the volumes of high-level radioactive, low-level radioactive, and transuranic waste generated by each of the alternatives under the proposed action. The volume of high-level radioactive waste to be disposed of in the repository under the proposed action would be reduced by 37 to 84 percent compared to the volume of spent nuclear fuel under the direct disposal option of the No Action Alternative. On the other hand, with the exception of Alternative 2, all alternatives under the proposed action would generate higher volumes of low-level radioactive and transuranic waste than the No Action Alternative.

All of the alternatives under the proposed action would either 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 tested and analyzed extensively under conditions relevant to a geologic repository. It is expected that other waste forms (e.g., ceramic, metallic) and possibly high-integrity cans not containing metallic sodium would be suitable for repository disposal.

Transportation

Incident-free transportation activities under the proposed action for alternatives not involving transportation of fuel to SRS would result in 0.000011 latent cancer fatalities among transportation workers (Alternative 6) and 0.0001 latent cancer fatalities in the total affected population (Alternative 6), compared to 1.2×10^{-6} latent cancer fatalities for transportation workers and 0.000011 latent cancer fatalities for the affected population under the No Action Alternative.

The latent cancer fatality risk to the affected population from postulated accidents under the proposed action and the No Action Alternative would be less than 1.0×10^{-10}, or one chance in 10 billion.

Incident-free transportation activities under the proposed action (for alternatives involving transportation of fuel to SRS) would result in 4.7×10^{-7} additional latent cancer fatalities among transportation workers and 6.0×10^{-6} additional latent cancer fatalities among the affected population over the duration of the transportation activities. These are risks from radiological causes. There would be approximately 0.00039 additional latent fatalities related to nonradiological (traffic) causes among affected urban populations along the transportation route.

The latent cancer fatality risk to the affected population from postulated transportation accidents would be 1.7×10^{-9}, or one chance in 588 million. The nonradiological (traffic) fatalities would be 0.0018.

S.9 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 is based on the incremental contribution from the maximum impacts from the proposed action added to baseline conditions at ANL-W and SRS and the maximum impacts from other onsite and offsite past, present, and other reasonably foreseeable future actions.

As indicated in Section S.8, the proposed action would have minimal or no impacts on land, visual, noise, geology and soils, ecological, and cultural and paleontological resources. The contributory effect from the proposed action on these resources would be so small that their potential contribution to cumulative impacts at ANL-W, INEEL, and SRS would be negligible.

Small but finite impacts have been identified for air and water resources and public and worker health and safety at ANL-W, INEEL, and SRS as a result of potential releases of radiological effluent and waste management activities. Treatment would transform the sodium-bonded spent nuclear fuel into another disposable waste form and small amounts of transuranic and low-level radioactive waste would be generated in the process.

The cumulative impact analysis in the EIS indicates that incremental impacts to public and worker health and safety from the proposed action, when added to the impacts from other past, present, and reasonably foreseeable future actions at ANL-W, INEEL, and SRS, would result in cumulative impacts well below applicable regulatory limits and guidelines. In addition, the projected amount of high-level radioactive, low-level radioactive, and transuranic waste generation, when added to the amount of waste generation from other past, present, and reasonably foreseeable future actions at these sites, would not require any additional treatment or storage capacities beyond the current and planned capacities at these sites.

Current and reasonably foreseeable future actions at ANL-W and INEEL include the Advanced Mixed Waste Treatment Program and the Idaho High-Level Radioactive Waste and Facilities Disposition Program, as described in the corresponding EISs. In addition, INEEL is also being considered as a possible site for the production of plutonium-238. This action and its cumulative effects will be addressed at a later date.

Reasonably foreseeable future actions at SRS include the Surplus Plutonium Disposition Program at SRS, the Interim Management of Nuclear Materials, the construction and operation of a Tritium Extraction Facility, the Surplus Highly Enriched Uranium Disposition Program, the management of plutonium and scrub alloy residues, and other DOE Complex miscellaneous components as described in the SRS Spent Nuclear Fuel Management EIS.

This page intentionally left blank.

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*
	Two Options Were Considered: **a. Continued Storage Until 2035,** **b. Direct Disposal Without Sodium Removal**	**Electrometallurgically Treat Blanket and Driver Fuel at ANL-W**	**Clean [a] and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W**
Air Quality - Radiological air emissions (curies per year)	Negligible impact 811 [c]	Negligible impact Tritium: 770 Krypton-85: 11,600	Negligible impact Tritium: 809 Krypton-85: 11,860
Water Resources - Radiological liquid effluent (curies per year)	No impact No liquid effluent	No impact No liquid effluent	No impact No liquid effluent
Socioeconomics	Loss of 350 direct jobs and 940 indirect jobs; no noticeable impact	Work force maintained; No impact	Work force maintained; No impact
Public and Occupational Health and Safety Risk			
• Project duration	35 years	13 years	9 years
• Normal operations (annual) [d]	**LCF [e]**	**LCF [e]**	**LCF [e]**
- Population	7.5×10^{-7}	1.4×10^{-6}	1.5×10^{-6}
- MEI	1.3×10^{-10}	1.7×10^{-10}	1.9×10^{-10}
- Average individual	3.1×10^{-12}	5.8×10^{-12}	6.2×10^{-12}
- Worker population	0.0088	0.0088	0.0088
- Average worker	0.000024	0.000024	0.000024
• Normal operations (project total) [d]			
- Population	6.5×10^{-6}	8.2×10^{-6}	8.3×10^{-6}
- MEI	1.1×10^{-9}	1.0×10^{-9}	1.0×10^{-9}
- Worker population	0.084	0.13	0.092
• Hazardous chemicals			
- MEI	None	None	None
Environmental Justice	No disproportionately high and adverse impact to minority or low-income populations		
Waste Management (cubic meters)			
• High-level radioactive waste	152 (Direct disposal volume) [f]	81.1	43.2 [g]
• Low-level radioactive waste	904	861	733.7
• Transuranic waste	12	14.1	10.7
Transportation Risk			
• Incident-free	**LCF [e]**	**LCF [e]**	**LCF [e]**
- Population	0.000011	0.000016	0.000016
- Workers	1.2×10^{-6}	1.8×10^{-6}	1.7×10^{-6}

LCF = Latent Cancer Fatalities; MEI = Maximally Exposed Offsite Individual.

[a] Clean means sodium removal.

[b] Over a period of six months.

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

[d] Annual value represents the maximum impact in a single year. Project total value represents the total impact to account for the variations in annual impact over the project duration.

Alternative 3		*Alternative 4*	*Alternative 5*		*Alternative 6*
Clean [a], Declad, and Package Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W	**PUREX Process Blanket Fuel at SRS [b]**	**Clean [a] and Melt and Dilute Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W**	**Clean [a], Declad, and Package Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W**	**Melt and Dilute Cleaned and Declad Blanket Fuel at SRS**	**Melt and Dilute Blanket and Driver Fuel at ANL-W**
Negligible impact Tritium: 809 Krypton-85: 11,860	Negligible impact Tritium: 162 Krypton-85: 1,187	Negligible impact Tritium: 809 Krypton-85: 11,860	Negligible impact Tritium: 809 Krypton-85: 11,860	Negligible impact Tritium: 54 Krypton-85: 396	Negligible impact Tritium: 2,162 Krypton-85: 32,650
No impact No liquid effluent	Negligible impact Tritium: 1.54 Other: less than 0.022	No impact No liquid effluent	No impact No liquid effluent	No impact No liquid effluent	No impact No liquid effluent
Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact
9 years	Less than 1 year	13 years	9 years	3 years	12 years
LCF [e]	**LCF [e]**	**LCF [e]**	**LCF [e]**	**LCF [e]**	**LCF [e]**
1.5×10^{-6}	0.000010	1.5×10^{-6}	1.5×10^{-6}	3.8×10^{-6}	6.1×10^{-6}
1.9×10^{-10}	2.6×10^{-10}	1.9×10^{-10}	1.9×10^{-10}	5.0×10^{-11}	1.0×10^{-9}
6.2×10^{-12}	1.2×10^{-11}	6.2×10^{-12}	6.2×10^{-12}	5.5×10^{-12}	2.6×10^{-11}
0.0088	0.015	0.0088	0.0088	0.02	0.0088
0.000024	0.0001	0.000024	0.000024	0.0002	0.000024
8.3×10^{-6}	0.000010	8.3×10^{-6}	8.3×10^{-6}	0.000011	0.000012
1.0×10^{-9}	2.6×10^{-10}	1.0×10^{-9}	1.0×10^{-9}	1.5×10^{-10}	2.0×10^{-9}
0.092	0.015	0.13	0.092	0.06	0.12
None	None	None	None	None	None
No disproportionately high and adverse impact to minority or low-income populations					
23.6 (18 at ANL-W; 5.6 at SRS)		63.6	94.62 (18 at ANL-W; 76.62 at SRS)		86
2,960.5 (770.5 at ANL-W; 2,190 at SRS)		845	1,178.5 (770.5 at ANL-W; 408 at SRS)		924
100.7 (10.7 at ANL-W; 90 at SRS)		12.8	27.2 (10.7 at ANL-W; 16.5 at SRS)		14.1
LCF [e]	**LCF [e]**	**LCF [e]**	**LCF [e]**	**LCF [e]**	**LCF [e]**
0.000015	6.0×10^{-6}	0.000072	0.000015	6.0×10^{-6}	0.0001
1.6×10^{-6}	4.7×10^{-7}	7.9×10^{-6}	1.6×10^{-6}	4.7×10^{-7}	0.000011

[e] Dose to the MEI, average individual, and the population (in person-rem) can be found by dividing the corresponding LCF values by 0.0005. Dose to the average worker and worker population (in person-rem) can be found by dividing the corresponding LCF values by 0.0004. The regulatory dose limit for offsite individuals (public) is 0.010 rem per person per year from air exposures, and 0.1 rem per person per year for all pathways. The administrative control limit for an individual worker at a DOE site is 2 rem per person per year.

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

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

S.10 GLOSSARY

Background Radiation — Ionizing radiation present in the environment from cosmic rays and natural sources in the Earth; background radiation varies considerably with location. The U.S. average background radiation is 300 millirem per year.

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 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 fissile 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 that (1) acts as a pour mold for solidified high-level radioactive waste, and (2) for spent nuclear fuel, may provide structural support for intact spent nuclear fuel, loose rods, nonfuel components, or confinement of radionuclides.

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 (e.g., melt and dilute product).

Curie — unit of radioactivity equal to 37 billion disintegrations per second; also a quantity of any nuclide or mixture of nuclides having 1 curie of radioactivity.

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

Depleted Uranium — Uranium with a smaller percentage of uranium-235 than the 0.7 percent found in natural uranium. It is a by-product of the uranium enrichment process, during which uranium-235 is collected from one batch of uranium, thereby depleting it, and added to another batch to increase its concentration of uranium-235.

Dilute — To reduce the concentration of a substance by adding another material to it.

Disposal — The isolation of radioactive waste from the accessible environment, as defined in 10 CFR 60.2. Disposal means the emplacement in a repository of high-level radioactive waste, spent nuclear fuel, or other highly radioactive material with no foreseeable intent of recovery, whether or not such emplacement permits the recovery of such waste.

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.

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 waste discharged into surface waters.

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

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

Fissium — An alloy of molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium.

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 sealed fissile material (fuel pin) 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 (1) the geologic repository operations area, and (2) the portion of the geologic setting that provides isolation. A near-surface disposal area is not a geologic repository.

Hazardous Waste — Any solid waste (also can 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 (1) highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid waste derived from such liquid waste that contains fission products in sufficient concentrations; and (2) other highly radioactive material that is determined, consistent with existing law, to require permanent isolation.

Latent Cancer Fatalities — Fatalities associated with acute or chronic environmental exposure to chemicals or radiation that occur from delayed effects years after 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 Offsite Individual — 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 potentially could receive the maximum dose of radiation or hazardous chemicals.

Metric Tons of Heavy Metal — Quantities of unirradiated and spent nuclear fuel are traditionally expressed in terms of metric tons of heavy metal (typically uranium), without the inclusion of other materials such as cladding, alloy materials, and structural materials. A metric ton is 1,000 kilograms, which is equal to about 2,200 pounds.

Millirem — One thousandth of a rem.

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

Normal Operation— All activities associated with a facility mission, whether operation, maintenance, storage, etc., 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 — Highly susceptible to spontaneous ignition and sustained combustion.

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

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

Roentgen Equivalent Man (rem) — A measure of radiation dose (e.g., 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.