
Volume 1: Executive Summary

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Performance Assessment of the Direct Disposal in
Unsaturated Tuff of Spent Nuclear Fuel and
High-Level Waste Owned by
U.S. Department of Energy

Volume 1: Executive Summary

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Abstract

This assessment studied the performance of high-level radioactive waste and spent nuclear fuel in a hypothetical repository in unsaturated tuff. The results of this 10-month study are intended to help guide the Office of Environment Management of the U.S. Department of Energy (DOE) on how to prepare its wastes for eventual permanent disposal. The waste forms comprised spent fuel and high-level waste currently stored at the Idaho National Engineering Laboratory (INEL) and the Hanford reservation. About 700 metric tons heavy metal (MTHM) of the waste under study is stored at INEL, including graphite spent nuclear fuel, highly enriched uranium spent fuel, low enriched uranium spent fuel, and calcined high-level waste. About 2100 MTHM of weapons production fuel, currently stored on the Hanford reservation, was also included. The behavior of the waste was analyzed by waste form and also as a group of waste forms in the hypothetical tuff repository. When the waste forms were studied together, the repository was assumed also to contain about 9200 MTHM high-level waste in borosilicate glass from three DOE sites. The addition of the borosilicate glass, which has already been proposed as a final waste form, brought the total to about 12,000 MTHM.

A source term model was developed to study the wide variety of waste forms, which included radionuclides residing in 10 different matrices and up to 8 nested layers of material that might react with water. Several important capabilities of this model, which is embedded into a two-phase composite-porosity fluid flow and transport model, are approximations of diffusion through rubble contacting a container, wet/dry oxic/anoxic corrosion, retardation on corrosion products, separate solubilities for inside/outside container, and corrosion rate dependence on temperature, presence of oxygen, water saturation, and time, if desired. In most cases, information readily available in literature was used to develop approximately 3700 data items needed by the models; when new data were required for the detailed phenomenological modeling used in this study, subjective estimates from analysts were necessary. The use of subjective estimates is in keeping with the need to provide immediate guidance to DOE on its waste forms, although improved defensibility of the input data would be desirable.
Abstract

The possibility and consequences of critical conditions occurring in or near containers of highly enriched uranium spent nuclear fuel were also studied. Although a full coupling of conditions that control a criticality occurrence was beyond the scope of this performance assessment, a range of temperature and water saturation was considered in preliminary reactor dynamics calculations, in which phenomena were manually coupled. Other analysis approaches included summarizing data from historical criticality accidents to bound probable conditions and performing scoping calculations in which the repository was assumed to act like a reactor at steady state.

The criteria used to measure performance included disposal system criteria, with special emphasis on the individual protection requirements, in the U.S. Environmental Protection Agency's Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Waste, 40 CFR 191 (although this standard has been set aside by the Energy Policy Act of 1992), and the waste package and engineered barrier requirements in the Nuclear Regulatory Commission's Disposal of High-Level Radioactive Wastes in Geologic Repositories, 10 CFR 60.

The general findings are as follows: For a repository in unsaturated tuff, disposal of currently existing DOE spent fuel and high-level waste, with minimal treatment or conditioning in a disposal container of carbon steel and Incoloy Alloy 825, behave similarly to commercial pressurized water reactor fuel and high-level waste in borosilicate glass. Furthermore, the waste complied with the overall dose and containment requirements of 40 CFR 191 over 10,000 yr. Technetium-99 was the major contributor to doses in the groundwater pathway in the first 10,000 yr, but neptunium-237 was the most important at 100,000 yr. The small inventory of carbon-14 in the DOE spent fuel helped in complying with the current containment requirements.

The temperature of the repository peaked at slightly above 90°C and stayed above 80°C for about 1000 yr when the containers were placed horizontally inside long tunnels with a power density of 23.5 W/m². Because of the high corrosion rate associated with the high temperatures, the disposal container did not meet the requirements of 10 CFR 60 even though the overall disposal system complied with 40 CFR 191. The differences in behavior among the waste types were partially caused by the local differences in temperature as the result of substantial differences in heat output per waste type. The cooler high-level waste performed better than the hotter spent fuels. No substantial difference was seen between the loose and immobilized calcine high-level waste currently stored at INEL.

Although the criticality calculations are preliminary, they support the belief that it would be difficult to create conditions that would provide enough water from infiltration in a 10,000-yr regulatory period to (a) corrode the containers, (b) remove neutron absorbers or uranium, and (c) moderate a nuclear chain reaction. Furthermore, should a criticality occur in or near a container in an unpressurized system at some future time, the nuclear dynamic calculations show that such a critical condition is extremely sensitive to minute changes in groundwater saturation that, in turn, are sensitive to changes in temperature. Therefore, it would be very difficult to keep a natural reactor operating at a power level similar to that which occurred with aqueous accidents. The calculations indicated that a noticeable increase in temperature lowered the saturation enough to stop the nuclear reaction. Finally, the bounding calculations, which assumed a steady-state criticality reaction similar to the low power of historical aqueous accidents, produces no more than 10²⁵ fissions operating over 10,000 yr. Thus, should all 289 containers with highly enriched uranium fuel go critical for 10,000 yr, only about 10²⁸ fissions occur, which is less than a 1% increase to the fission products (radionuclide inventory) already existing in a 12,060-MTHM repository, given an average burnup of the spent fuel.
Preface

This document reports on a performance assessment for the US Department of Energy (DOE). The general purpose of this study is to assist the DOE in determining the proper direction for its technology development program for spent nuclear fuel and high-level waste currently stored at the Idaho National Engineering Laboratory (INEL) and elsewhere. Although based on preliminary data, the results of this performance assessment are intended to aid decisionmaking regarding what types of waste treatment and packaging should be used for DOE waste. These decisions must be made by DOE in response to legal requirements for immediate treatment and removal of waste at specific sites.

To achieve its general objective, the 1994 performance assessment studied the effects of the disposal of spent nuclear fuel and high-level waste under the following conditions: minimal treatment of the spent fuel and waste, use of containers with layers of carbon steel and a corrosion-resistant material, and disposal in a geologic repository in unsaturated, volcanic tuff. The spent fuel under study originated in military and experimental reactors, and the high-level waste was generated during reprocessing of the spent fuel. Additional conditions studied in this performance assessment are the probability and consequences of a critical condition developing in a closed and sealed repository some time after waste disposal.

Approaches Adopted by the 1993 and 1994 Studies on Unresolved Policy Issues

Although the United States Congress is pursuing a policy for waste disposal based on the Energy Policy Act of 1992 (Energy Policy Act, 1992), not all policy issues have been resolved. Four policy issues that are not final but are relevant to the overall study are (1) establishing a location for disposal of the DOE waste, (2) establishing performance criteria, (3) determining the usefulness of waste treatment versus robust containers, and (4) determining the acceptability of a criticality in a deep, geologic repository.

The approach taken with the first issue, establishing a location for disposal of the DOE waste, was to examine more than one geologic medium to determine how different media might affect the need for waste treatment. In the initial assessment performed last year (herein called the 1993 PA), salt and granite disposal systems were studied. For this year’s study (the 1994 PA), the medium was unsaturated, volcanic tuff.

Because of changes in policy, no fixed regulations currently exist. Thus, the approach to the second issue was to use criteria from regulations that have been used in past studies, although these regulations are currently undergoing revision, and also to include criteria in anticipation of future regulations. For example, this study includes calculations on doses to individual members of the public over time.

The third issue concerns whether treatment options for waste should be pursued or whether the emphasis should be placed on a robust container. In the 1993 PA, several alternative treatments were evaluated with a simple, thin-walled handling container. A minimal treatment option with a robust container was studied in the 1994 PA. The emphasis on the robust container meant that the assumptions about its corrosion resistance and the amount of fluid available were particularly important; thus a significant effort was extended to further improve the source term model for the 1994 PA. Although the 1994 PA was originally intended to examine the same treatment options as the 1993 PA, receipt of funding for phase 2 late in the fiscal year limited Sandia’s ability to provide a study of unsaturated tuff that included all conditions of the 1993 PA while at the same time enlarging its scope. In acknowledgement of these time and funding constraints, Sandia selected only one treatment option for spent fuel—minimal treatment. This decision was based on a result from the 1993 PA that indicated not only that all five of the treatment options potentially complied with the regulatory criteria under study, but also that only the extensive treatment improved performance.

The fourth issue—the acceptability of a criticality in a repository—was approached in the 1993 PA by examining the implications of preventing any criticality by limiting the fissile mass per container. In the 1994 PA, a set of computations was developed to evaluate the implications of a critical event occurring in a deep, geologic repository.
The concrete vaults do not meet the current requirements for secondary containment, because acidic waste solutions requires planning among the State of Idaho, the Environmental Protection Agency (EPA), and other government entities. A decision must be reached either to build new facilities (e.g., a new calcine facility) or modify existing basins that do meet current standards such that they can hold more fuel. Either decision could alter the concrete if a leak were to develop. Thus, these vaults are not in compliance with the EPA regulations and the resulting high-level wastes. Since 1963, the Chem Plant has processed the liquid high-level wastes and some hazardous wastes to form a dry granular power, called "calcine." Currently, 3800 m$^3$ of calcine, 1900 m$^3$ of liquid waste ready for calcination, and 5700 m$^3$ of decontamination, sodium-bearing waste, which requires further treatment before calcination, are stored at INEL. In addition, the decision in 1992 by the national administration to discontinue reprocessing of spent nuclear fuel has left 768 MT of spent fuel in storage with unspecified plans for future disposal.

Legal issues as the result of this situation include (1) disposition of currently stored spent fuel, (2) disposition of calcined waste, and (3) removal and treatment of high-level and hazardous wastes stored in stainless steel tanks.

The first issue pertains to a portion of the spent nuclear fuel that is stored in aging basins, which are not lined and thus not built to today's standards. A decision must be reached either to build new facilities (e.g., a dry fuel storage facility) or modify existing basins that do meet current standards such that they can hold more fuel. Either decision requires planning among the State of Idaho, the Environmental Protection Agency (EPA), and other government entities.

The second item is for DOE to proceed with its good-faith negotiations to find or build treatment facilities for all its wastes so that the State of Idaho will continue to consent to operate the calcine facility and store the calcine. Calcine is a listed, mixed waste and is subject to the storage and land disposal restrictions of the Resource, Conservation, and Recovery Act (RCRA).

The third item, which has the highest legal priority of these items, is to meet existing milestones in a consent order to remove and treat the high-level and hazardous wastes stored in stainless steel tanks within concrete vaults. The concrete vaults do not meet the current requirements for secondary containment, because acidic waste solutions could alter the concrete if a leak were to develop. Thus, these vaults are not in compliance with the EPA regulations that implement RCRA.

Although these three items are of immediate concern because of their legal implications, the DOE initially sought guidance for the long term about ways to prepare the waste (such as treatment, conditioning, and packaging) for eventual disposal. The program chose a performance assessment as the means of providing technical input regarding this issue. Current financial demands as a result of the legal commitments, however, have curtailed funding for the long-term studies.
Organization of Report

The executive summary (Volume 1) reviews the scope and method of analyses, but its emphasis is on the important ideas and conclusions of the 1994 performance assessment. Volume 2 is a detailed account of the analysis design and conclusions. The introductory chapter describes the purpose of the report, briefly describes the regulatory setting, sets the stage for the data and concepts that are presented in later chapters, and highlights some of the analysis decisions made during the assessment. Chapters 2 and 3 provide background information on the legal setting for nuclear waste disposal and the performance assessment process. Specifically, Chapter 2 provides details on the current national policy (Energy Policy Act of 1992); on the most recent version of U.S. Environmental Protection Agency (EPA) Standard, *Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes*, 40 CFR 191, which is to be revised; and on the unmodified version of the Nuclear Regulatory Agency’s *Disposal of High Level Radioactive Wastes in Geologic Repositories*, 10 CFR 60. Chapter 3 gives an overview of the probabilistic concepts and the general theoretical steps of a performance assessment process.

The remaining chapters cover the tasks of the performance assessment in the order in which they are described in the Introduction, i.e., disposal system characterization (Chapters 4, 5 and 6); scenario development (Chapter 7); probability modeling (Chapter 8); consequence modeling (Chapters 10 through 14); and results (Chapter 15). The consequence modeling task is grouped in chapters by topic to help the reader locate material of particular interest, specifically, direct releases through cuttings (Chapter 9), criticality modeling (Chapter 10), radionuclide source term modeling (Chapter 11), groundwater flow and transport modeling for the Complex PA (Chapter 12) and the Simple PA (Chapter 13), and biosphere modeling (Chapter 14). Because of the extensive amount of information that must be described for each major task of a performance assessment, the chapters primarily summarize or highlight models and data that are critical or important to the performance assessment. Additional information necessary to provide a complete picture of the disposal system is provided in the appendices (Volume 3 of this report).

Although the report documents both the method of analysis and the data used, the organization of the report emphasizes the method of analysis. Therefore, the data are not isolated, but instead are found in related chapters. Most model parameters for characterization of the disposal system are included in Chapters 4, 5, and 6. The parameters for gas generation/consumption and radionuclide source term are detailed in Chapter 11, which describes these submodels.

Suggested Use of Report

- For a summary of the results, refer to the Executive Summary (Volume 1).
- For an overview of the entire performance assessment process, refer to Chapter 1, Introduction (Volume 2).
- For specific details pertaining to performance assessment tasks, consult the individual chapters (Volume 2).
- For background information pertaining to specific performance assessment tasks, consult the appendices (Volume 3).

Related Documents

This performance assessment is the second phase of a study that was begun in 1993. The earlier study examined the effects of five waste treatment options for spent nuclear fuel and high-level waste after disposal in hypothetical salt and granite repositories. The performance assessment was documented in the following report:


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The major source of data for site characterization of the tuff repository was drawn from Sandia's performance assessment* of the Yucca Mountain Project:


In addition to the site characterization data, the 1994 PA also used the unsaturated flow and transport code, TOSPAC, from the Sandia Yucca Mountain Project in some calculations.

Both the 1993 and 1994 PAs use the performance assessment methodology developed for the study of the Waste Isolation Pilot Plant (WIPP), near Carlsbad, NM. Thus, a useful companion document that reviews the mechanics of the performance assessment process is


Analysis Rigor

The work reported herein (software, analysis, and data) is, for the most part, experimental or preliminary. In general, most initial performance assessments are carried out at this level because information about the disposal system under study is not completely known; analyst judgment must supplement the available information. In terms of quality assurance, the class of data, software, and analysis used in this study is Level X (experimental). Although this level of quality assurance is not a reflection of the intrinsic value of the work, it does indicate that an ability to trace back to records such as adjudicated sources for the data, current user and theory manuals, and planning documents of the analysis, cannot be guaranteed. However, care was exercised in the three main areas affecting the results: data, software, and analysis. For example, most of the model parameters for the geologic barrier came from a previous performance assessment, as noted; some verification of the software has occurred (Appendix G); and verification that the software has been properly applied in the analysis was examined both internally among the analysts, since numerous applied models were developed for the stochastic simulations, and also externally by a technical peer panel and a program review panel.

These quality assurance efforts have attempted to ensure that the 1994 PA represents a reasonable undertaking that includes, implements, and evaluates important phenomena, pertinent data, and the general state of knowledge regarding relevant issues. As previously noted, this performance assessment is not intended to serve as support for licensing an unsaturated tuff repository for nuclear waste.

A complete description of the quality assurance levels, which were originally developed for the Waste Isolation Pilot Plant Project for the 1989-1992 performance assessments, and the assurance that each provides regarding traceability, repeatability, peer review, and other issues, can be found in the following reports:


* The Performance Assessment Departments at Sandia for the Yucca Mountain Project (YMP) and the Waste Isolation Pilot Plant (WIPP) Project are both within the Nuclear Waste Management Center. Their style of calculations differ because of the stage of the projects, degree of scientific understanding of unsaturated flow, and specific requests from their respective DOE sponsors. Although the 1994 PA reported here is a product of the WIPP Performance Assessment Department, M.L. Wilson, a member of the YMP Performance Assessment Department, participated as a member of the PA team.
When future performance assessments examine related areas and concepts, with more accurate data and a better understanding of the important phenomena to study, the findings of this study may be modified. Therefore, decisions based on this performance assessment should encompass the knowledge that the findings may be adjusted in the future and, therefore, that impacted decisions might also be re-examined.

Acknowledgments

The authors of this report appreciate the review of the performance assessment approach and its application provided by the technical review panel, established by Sandia, which includes the following individuals:

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Karsten Pruess, Lawrence Berkeley Laboratory, Berkeley, CA
Richard Westerman, Battelle Pacific Northwest Laboratories, Richland, WA.

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J. M. Boak, Chemical Science and Technology, Los Alamos National Laboratory,

who participated in the program review meeting, established by LITCO, and the reviews by H. H. Loo and his staff at LITCO. Finally, the editorial help on the text and illustrations provided by J. Chapman and D. Pulliam, respectively, of Tech Reps, Inc., Albuquerque, NM, greatly improved the report.

A performance assessment of a complex system using stochastic simulation requires a tremendous volume of long running calculations; thus, the reliability of the computer system is very important. The help of M. McNeil and J. Geilow of the Computational Support Department is acknowledged.

Description of Participants

The primary coordinator for the study reported here is Lockheed Idaho Technologies Co., Inc. (LITCO), which recently took over operation of Idaho National Engineering Laboratory from the Westinghouse Idaho Nuclear Company (WINCO), under which this work was initiated. Lockheed Idaho Technologies Company characterized the spent fuel and high-level waste, and supplied the waste container design concepts, which were adapted from the most recent design concepts being explored by the Yucca Mountain Project. They also performed a preliminary criticality safety analysis, as part of a team effort to explore the likelihood and consequences of a critical condition in or near a waste container. The Waste Isolation Pilot Plant (WIPP) Performance Assessment Department at Sandia National Laboratories, a major team member for this program, provided expertise in performance assessment methodology and is the primary author of this report.
A performance assessment is a multidisciplinary task and requires the expertise of numerous individuals. The individuals who have contributed to this 1994 (both full- and part-time) are noted in the organization chart that follows.

Organization Chart for INEL Project (October, 1994)

Managers

- INEL System Analysis
  - R. Klingler, LITCO
- Sandia
  - PA Manager: D.R. Anderson, SNL
- INEL PA Analysis
  - H. Loo, LITCO
- INEL Project Task Leader: R.P. Rechard, SNL

Activities and Subtask Leaders

- Waste Characterization
  - (Ch. 4, App. G, and App. F)
  - J.S. Rath, NMERI
  - H. Loo, LITCO

- Repository Design (Ch. 6)
- Cutting Modeling (Ch. 10)
- Analysis Toolbox Enhancement
  - J.S. Rath, NMERI

- Geologic Barriers (Ch. 5)
- Secondary Data Base
  - K.P. Brinster, SAIC

- PA Overview (Ch. 3)
- Probability Model (Ch. 9)
- Scenario Development (Ch. 17)
- Gas Transport Interpolation (Ch. 19)
  - M.S. Tierney, SNL

- Room Chemistry (Ch. 11)
  - C.T. Stockman, SAIC

- Complex Repos. Models (Ch. 12)
- Model Verification
- Repository Modeling
  - J.L. Ramsey, SNL

- Scoping PA Calculations (App. E)
- Simple PA Models (Ch. 13)
- Flow & Transport
  - M.L. Wilson, SNL

- Complex Aquifer Flow & Transport (Ch. 12)
  - D.M. Stoelzel, SNL

- Biosphere Transport
  - R.D. McCurley, NMERI

- Sensitivity Analysis (Ch. 15)
  - L. Smith, SAIC

- Compliance Issues
  - (Ch. 2, App. C, and App. D)
- Review Panels
- QA Coordination
- Management Support
  - K.M. Trauth, SNL

- Document Support
  - J. Chapman, TRI

- Criticality Conditions
  - (Ch. 10, App. B)
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  - D.R. Evans, LITCO

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- D.K. Rudeen, NMERI
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- D.H. McCurley, SNL
  - B. DeLapp, SNL
  - G. Pullen, SNL
  - G. Tuller, TRI

- M.M. Gruebel, TRI
  - D.L. Pulliam, TRI
  - L. Tartaglia, TRI

- J.R. Wilson, LITCO
  - P.J. Santer, LITCO

API - Applied Physics Incorporated
GeoC - Geo-Centers Incorporated
NMERI - New Mexico Engineering Research Institute
SAIC - Science Applications International Corporation
SNL - Sandia National Laboratories
Spr - Spectra Research Institute
TRI - Tech. Reps., Incorporated
LITCO - Lockheed Idaho Technologies Company
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Executive Summary

R. P. Rechard

The purpose of this report is to assist the Office of Environmental Management of the U.S. Department of Energy (DOE) in evaluating the feasibility of disposing its spent nuclear fuel and high-level waste with minimal treatment in a geologic repository in tuff. The study was developed in response to a request for information from the DOE regarding its technology development program for spent nuclear fuel and high-level waste currently stored at the Idaho National Engineering Laboratory (INEL) and elsewhere.

A performance assessment was chosen as the means of providing this information not only with regard to a long-term perspective on waste disposal decisions but also to affect immediate concerns about waste treatment and packaging in response to court orders and other legal requirements. In addition to providing guidance to the project, this performance assessment serves to inform both the analysis team and decisionmakers who commissioned the study about (a) the type of model development and analysis that should be conducted in the future and (b) the scientific information that must be gathered, either from the literature or investigative studies, for use in models selected for future performance assessments. Questions that were addressed included what decisions about waste preparation could be made before a repository site had been identified and what decisions should be made only after a repository site was known. This evaluation may also lead to the development of minimum acceptance criteria for disposal of the waste in a generic, deep geologic repository and may assist the DOE in developing and demonstrating the capability for nuclear waste disposal.

In this study (herein referred to as the 1994 PA), the performance of containers of radioactive materials, when placed in an unsaturated tuff repository, is examined for compliance with US regulations. Dose calculations were also performed. The environmental criteria considered in this performance assessment are those of the Environmental Protection Agency (EPA) standard, Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes, 40 CFR 191 (EPA, 1985; 1993), which uses a 10,000-yr period, and those of the Nuclear Regulatory Agency (NRC) regulation, Disposal of High-Level Radioactive Wastes in Geologic Repositories, 10 CFR 60 (NRC, 1983). The EPA standard, 40 CFR 191, was chosen because it is being applied at the Waste Isolation Pilot Plant, which is a repository in bedded salt for waste contaminated with transuranic radionuclides from the DOE's nuclear weapons complex. The NRC's 10 CFR 60 was chosen because it contains strict requirements for providing substantially complete containment within the emplacement package for 300 yr and limits on maximum release rates from the engineered barrier thereafter.* In anticipation of changing regulations, individual doses were also considered in this performance assessment based on individual protection criteria from 40 CFR 191 and current international studies.

The hypothetical repository for the 1994 PA in unsaturated tuff is similar to the potential repository at Yucca Mountain, Nevada.† A tuff disposal system is being modeled for the 1994 PA not only to compare its results to those from the generic salt and granite disposal systems studied earlier (1993 PA), but also to examine the effects of disposal of DOE spent fuel in a Yucca Mountain-like repository. The materials considered for disposal are spent nuclear fuel and high-level waste. The spent fuel includes graphite fuel, experimental highly enriched fuel, robust highly enriched fuel, low enriched fuel, and fuel for producing weapons material. The high-level waste is calcine, which is generated during reprocessing of spent fuel. These waste types are analyzed (1) individually by type and (2) in combination as one disposal group. The latter analysis, which is designed to simulate more realistically the performance of a repository with more than one waste type, also includes defense high-level radioactive waste immobilized in borosilicate glass. Results from both analysis methods are presented in this summary.

*At this time, it is not clear what the environmental requirements will be for an actual geologic repository for high-level waste and spent nuclear fuel. Currently, the National Academy of Sciences, at the request of the EPA and based on requirements in the Energy Policy Act of 1992 (Public Law 102-486), is exploring possibilities for the development of a new standard to replace 40 CFR 191 for the proposed repository for high-level waste and spent nuclear fuel at Yucca Mountain, Nevada. This action also affects the NRC regulation, 10 CFR 60. See Volume 2, Chapter 2 for more discussion.

†At present, the only potential site in the United States for a geologic repository for high-level waste and spent nuclear fuel is at Yucca Mountain, Nevada, in unsaturated tuff. Analysis of this site was not pursued in the initial performance assessment reported last year (herein referred to as the 1993 PA) because of the technical challenges presented by modeling an unsaturated tuff repository and because bedded salt and granite were thought to adequately bracket the behavior.
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Modeling of complex phenomena for the 1994 PA was incorporated directly into the stochastic performance assessment. The models included a source term model that could accommodate the diverse fuel types; a two-dimensional fluid flow and transport model incorporating two-phase flow and composite porosity; and a three-dimensional saturated/unsaturated, dual-porosity model. The 1994 PA also studied the possibility and consequences of critical conditions developing in one or several containers of highly enriched uranium spent fuel. Although a full coupling of conditions that control a criticality occurrence was beyond the scope of this performance assessment, ranges of temperature and water volume were considered in preliminary reactor dynamics calculations.

This summary presents a brief description of the analysis methodology (Section ES.1), emphasizes the general findings of this ambitious performance assessment (Section ES.2), and provides guidance for the Spent Nuclear Fuel and High-Level Waste Management Technology Development Program at the Idaho National Engineering Laboratory, of which this study is a part (Section ES.3).

ES.1 Analysis Design

The analysis design for the 1994 PA is summarized in Volume 2, Chapter 1, of this report. Detailed discussions of the design components make up the major portion of that volume, including analysis approach, waste parcels considered, and analysis assumptions. The overview presented below, which also briefly covers these topics, is intended simply to provide a framework for the reader so that he or she can obtain a measure of understanding of the findings and recommendations presented in Sections ES.2 and ES.3.

As stated previously, the focus of this analysis is to help guide the DOE in its decisions about how to prepare high-level wastes and spent fuel stored within the DOE complex for eventual permanent disposal. Specific goals were to (1) evaluate the potential impacts of minimal treatment on a compliance determination of spent fuel and high-level waste for disposal in an unsaturated tuff geologic repository (Section ES.2), (2) include effects of critical conditions in or near a waste container (Section ES.2), and (3) provide information about the current status of available models and data, which could be used in developing minimum waste acceptance criteria and guidance for technology development programs (Section ES.3).

To meet these specific goals, the approach for this analysis was to (a) use the methodology established by Sandia National Laboratories for assessing the long-term performance of a geologic disposal system, (b) use data collected from past assessments of an unsaturated tuff disposal system, (c) examine several levels of simplification in the consequence models to develop an understanding of the uncertainty of the results, (d) use several analysis approaches to evaluate the probability and consequences of a self-sustaining nuclear chain reaction in or near a container to develop a protocol for future repository criticality studies, (e) consider a minimal waste treatment option in a robust container, and (f) develop a detailed source term model. Section ES.1.1 discusses items a through e; caveats regarding the analysis are presented in Section ES.1.2. The minimal treatment option and the source term model are discussed in Section ES.2. Recommendations about model development, analysis, and data gathering are provided in Section ES.3.3.

ES.1.1 Analysis Approach

Summarized below are descriptions of the analysis methodology, general sources of data, levels of model simplification, criticality calculations, and possible benefits of future iterations of the performance assessment.

For discussion in the 1994 PA, the simulation process is categorized into six general steps: (1) collection of data on spent fuel and high-level waste, geology and hydrology near the site, and facility design; (2) identification of features and agents of the disposal system whereby radionuclides might be released to the accessible environment and the selection of what to model and the methodology; (3) development and execution of probability models to determine the uncertainty in model parameters and predict the likelihood of the scenarios; (4) development and execution of consequence models to predict the amount of radionuclide release, including evaluating the uncertainty associated with these predictions; (5) development and execution of criticality models, analysis of criticality calculations, and predictions of criticality occurrence; and (6) consideration of other issues that affect the disposal repository, the environment, and the potential for criticality.

The waste parcel is defined as the waste form (i.e., treated waste type), handling container, and disposal container(s) (overpack).

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with the predictions; (5) comparison of the predicted releases and doses with regulations and guidelines; and (6) evaluation of important variables that influence the results (Figure ES-1).

Methodology. The primary means of determining whether waste is acceptable for geologic disposal is through a process called a performance assessment. In general, a performance assessment is an analysis that assesses whether a system meets a set of performance criteria. For the analysis documented in this report, the system is actually a composite mathematical model that represents a deep geologic repository disposal system for spent nuclear fuel and high-level waste. The performance criteria are long-term (10,000 yr) metrics (not short-term operational safety issues) specified by U.S. government regulations. Thus the approach, of necessity, is intimately tied to the process of building scientific models. In essence, the performance assessment can be viewed as a very complex process that estimates various performance metrics through models for comparison with applicable regulations.

The 1994 PA incorporates a stochastic simulation to present the results in probabilistic terms, specifically as a complementary cumulative distribution function (CCDF). In addition, the 1994 PA methodology includes a variety of approaches in its study of criticality concerns. A stochastic simulation is used because many uncertainties that are inherent in the modeling effort can be expressed probabilistically. The performance assessment procedure, while differing in its practical application, is identical to that of a probabilistic risk assessment (PRA) for a nuclear reactor. This methodology (which was also used in last year's performance assessment, referred to herein as the 1993 PA) is essentially the same as that used by Sandia for assessing the proposed Waste Isolation Pilot Plant (WIPP) disposal system near Carlsbad, New Mexico. However, at the WIPP, the methodology is being performed primarily to assess performance for evaluating compliance with regulations, whereas the results of the 1994 PA are intended to be used as a basis for decisions on resource allocation by DOE.

Collection of Data. The Lockheed Idaho Technology Company (LITCO) provided the best data available in July, 1994, on the inventory, geometry, and condition of spent nuclear fuel and high-level waste owned by the DOE. The information on the waste containers was obtained by LITCO by means of personal communication with members of the Yucca Mountain Project, which LITCO then modified for the purposes of this study. Data on the behavior of this waste and containers in a tuff environment, which was necessary for the source term model developed for the 1994 PA, was obtained by literature review and analyst judgment at Sandia. The data on the geology and hydrology of the tuff disposal system was obtained from Sandia's most recent evaluation of the proposed repository at Yucca Mountain (Wilson et al., 1994). The concept on facility design was developed by means of personal communication with Sandia staff working on the Yucca Mountain Project and then greatly simplified for the purposes of this study. About 3700 data items were collected and stored in a database for use in the 1994 PA.

Scenario Development. A few general scenarios were identified and used in the 1994 PA. Features selected include a collection point at the surface where precipitation gathers; a highly fractured, pancake stratigraphy; a large unsaturated zone in which the repository is located; vertical and horizontal emplacement of waste containers; and an underlying aquifer. The primary event considered is inadvertent human intrusion but, like the 1993 PA for the granite disposal system, the permeability of the fractures in the tuff was assumed to be high enough that additional fluid paths created by the drilled boreholes would not significantly enhance fluid movement and radionuclide transport.

The processes considered are two-phase flow with heat conduction and convection in a fractured porous matrix, large degrees of sorption of selected radionuclides on tuff, infiltration variations from climatic change, radiative heat transfer through the air gap between the container and tuff rock, gas consumption from oxic corrosion, control of radionuclide release as the result of localized corrosion, diffusion through container breach into tuff rock, degradation of cladding and matrix that hold radionuclides, and control of radionuclide release by either solubility, matrix alteration, or immediate release of some radionuclides in gaps and grain boundaries of spent fuel.

The events associated with developing critical conditions in a repository, a previously unexamined topic, were the result of a joint effort by LITCO and Sandia.

** As noted earlier in this summary, the environmental regulations for disposal sites in the United States are currently undergoing revision and so perhaps only the internationally recognized criterion of individual doses can be easily identified at this time.

† In general, a PRA is associated with only engineered facilities over short time scales, while a PA is associated with a combination of natural and engineered systems over geologic time scales. These relationships account for most of the practical differences between a PRA and a PA.

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Figure ES-1. Sequence of performance assessment analysis and corresponding organization of the main report (Volume 2). The approximate number of runs for the final analysis are also shown.
Probability Modeling. The probability of the inadvertent human intrusion into the repository by exploratory drilling was evaluated with a homogenous Poisson probability model. The probability of critical conditions occurring in or near a waste container was evaluated in a fault tree with subjective estimates of events leading to critical conditions. The probability distributions for the imprecisely known model parameters were developed by means of several techniques. For instance, the source term model distributions were often subjective estimates based on literature, while some model parameters of the geologic barrier were based on direct measurements.

Consequence Modeling. The 1994 PA examined the performance of minimally treated waste forms, which relies on the characteristics of the waste as is and the engineered design of the container. Thus, two efforts of prime importance in the 1994 PA were modeling phenomena related to (a) the release of radionuclides and (b) the consequences of a critical condition occurring in a closed and sealed repository some time in the future.

Source Term Modeling. The source term model was developed to accommodate the diverse characteristics of the waste. Capabilities of the model included tracking of full radioactive decay for an unlimited number of radionuclides; accommodation of waste package layers, including radionuclides residing in 10 different matrices and up to 8 nested layers of material; approximations of diffusion through rubble contacting a container; wet/dry oxic/anoxic corrosion; retardation on corrosion products; separate solubilities inside/outside container; and corrosion rate dependence on temperature, presence of oxygen, water saturation, and time, if desired.

Model Simplification for Base Case and Inadvertent Human Intrusion. In an engineering analysis, physical phenomena (e.g., processes in the repository and host rock) are simplified. Although the level of simplification that is appropriate cannot always be known a priori, evaluating the appropriate level is an important task. For this study, two distinct levels of simplification of the consequence models of the tuff disposal system were performed in parallel: (1) calculations to assess disposal system performance for individual waste types using sophisticated, lumped-parameter or one-dimensional models (Simple PA), including some models from past assessments by the Yucca Mountain Project, and (2) calculations to assess disposal system performance for a mixed waste group using models based on spatially distributed parameters (Complex PA). The models for the Complex PA are extensively modified versions of those used to evaluate the salt and granite disposal systems of the 1993 PA. The general findings from both the Simple and Complex PAs are presented in this summary. Figure ES-1 illustrates the sequence of calculations performed in parallel. The number of stochastic simulations performed in the 1994 PA range from 38 runs for the criticality consequence model to about 1100 runs for the Simple PA (Figure ES-1).

Model Simplification for Criticality Calculations. The 1994 PA studied the possibility of a critical condition occurring in or near a container of highly enriched uranium spent fuel. Within the performance assessment methodology, a critical condition was treated as an event and an individual scenario was constructed. Characteristics of a criticality were developed from information on characteristics of the tuff disposal system, spent nuclear fuel, and past experiments, accidents, and natural reactors. From this information, the features, short-term phenomena, and long-term phenomena of the criticality were identified.

The probability and consequences of the criticality scenario were examined in two ways. One method employed a traditional criticality safety analysis that followed the steps of a probabilistic risk assessment. In this analysis, a fault tree, based on hypothesized features and short- and long-term phenomena, was used to estimate probabilities, and bounding calculations were run to evaluate consequences; this analysis was performed by LITCO. The second approach evaluated whether the hypothesized features and short- and long-term phenomena would likely influence or promote critical conditions, and also included detailed calculations to evaluate the consequences more precisely. This latter analysis was performed by Sandia. Both approaches were performed in parallel so that the task could be completed within a 10-month period. The general findings from both types of analyses are presented in Section ES.2.

Comparison with Regulations and Sensitivity Analysis. The results from evaluating the entire disposal system are displayed as complementary cumulative distribution functions (CCDFs) for comparison with criteria in 40 CFR 191. These results include the releases integrated over 10,000 yr and then normalized by the EPA limit, and the doses to a maximally exposed individual from the aquifer below the repository. The individual doses were probabilistic for the first 10,000 yr. However, two deterministic runs using the mean and median values of the model parameters were extended to 100,000 yr. In addition, the mean breach times and release rates from the waste package and engineered barrier, respectively, are compared to the criteria in 10 CFR 60.
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Performance Assessment Iterations. Generally, a performance assessment is iterative, with each iteration covering different areas or exploring new concepts while reevaluating results of previous assessments. In this way, the system under study becomes better understood. Because this performance assessment is the first iteration in unsaturated tuff, neither the data underlying the model parameters nor the model form is fixed; all aspects of this study are subject to review and subsequent change in future iterations. (Additional caveats are provided in Section ES.1.2.) This report provides an opportunity for interested parties to consider some of the options that the DOE may explore in the future and to give constructive input for additional performance assessments.

ES.1.2 Analysis Caveats

Although the detailed assumptions cannot be summarized here, Volume 2 attempts to list clearly all major assumptions about the tuff disposal system. In fact, the model description and assumptions comprise the major portion of Volume 2. Five general caveats regarding the analysis are as follows.

First, this report was commissioned to provide information to the DOE concerning guidance about decisions on program direction and is not intended as support for a license application. In this situation, even preliminary findings from a performance assessment using preliminary data, such as described herein, are extremely pertinent and helpful in making reasonable choices. Hence, the general results are presented in this executive summary with the understanding that the reader appreciates that the findings are conditional on the models and data used to generate them. The models of the disposal system must be judged in combination with the results to set priorities for additional data acquisition, model development, and other decisions on general program direction. To make these judgments, details concerning the models and the data must be known. These detailed assumptions should be examined before actions are taken on the findings and the guidance presented in this summary.

Second, this is an initial performance assessment of DOE spent fuel and calcine high-level waste in unsaturated tuff in which information is being gathered to understand the behavior of this complex disposal system. The underlying data are based on the best information currently available, but should still be considered preliminary.

Third, although a large amount of work was accomplished, examination of critical conditions in the repository is preliminary. The preliminary steady-state calculations suggest that the effects of a criticality (the equivalent of a low power reactor operating continuously for 10,000 yr) in a repository would be negligible (less than 1% increase in fission product inventory), and preliminary dynamic reactor calculations cannot easily demonstrate the feasibility of operating a natural unpressurized reactor. Providing a defensible technical position on criticality in a geologic repository, however, requires further analysis that involves analyzing other criticality situations as outlined in Section ES.3.3.

Fourth, the environmental regulations for disposal sites in the United States are currently undergoing revision. This assessment primarily used the performance criteria in the Containment Requirements of 40 CFR 191 (§191.13) and the engineered barrier performance criteria of 10 CFR 60 (§60.113). In addition, calculations were performed to determine human doses after radionuclides were transported from a drillhole into a well of drinking water located beyond the controlled area in the accessible environment.

Fifth, the assessment of the disposal system is performed on a model of the disposal system, not the system itself. Because the U.S. regulatory criteria were also developed using simplified models of generic disposal systems, comparing system-generated data to the regulatory criteria is appropriate. However, the generally accepted style (e.g., the level of model detail expected and accepted assumptions of events such as inadvertent human intrusion) of these types of compliance calculations continues to evolve over time.

ES.2 Findings from the Analysis

This performance assessment uncovered many pieces of information about the waste form and its interaction with the disposal system. In many cases, the performance assessment analysis identified important parameters (i.e.,
data that are critical for determining how the system will perform over the long term). In some cases, the analysis
pointed to data that are considered valuable but which are either limited or not currently available. Although the find-
ings encompass the interaction of the components of the disposal system, organizing the results by component is use-
ful. Hence, the acquired information and modeling issues are categorized into seven areas: general findings about
the system and then specific findings about the waste form, the waste containers, the repository, the geologic barrier,
the agents acting on the disposal system, and critical conditions. These conclusions are presented in the following
sections as a summary of Volume 2, Chapter 15, “Results and Conclusions,” which contains information supporting
these findings.

ES.2.1 General Findings

When a site applies for a license, the absolute position of the results with relation to the total system criteria of 40
CFR 191 (or replacement regulations) is crucial. In the 1994 PA, the results indicate compliance with the perfor-
mance measures from 40 CFR 191, i.e., EPA summed normalized releases and individual dose. Table ES-1 lists these
findings, and Figure ES-2 provides support for some of these findings. However, because much of the data used in
the analysis is preliminary, there is a large uncertainty associated with the absolute position. Thus, the value of show-
ing this absolute position here is to recommend areas in need of improvement with regard to future licensing.

Table ES-1. Summary of General Information Gathered on the Total Disposal System Performance

1. Mean individual doses from a farm family using well water from the aquifer underlying the repository at (a) the
disposal system boundary (5 km from waste) and (b) 2.4 km from waste are too small to accurately calculate
after 10,000 yr.
   - The CCDF of mean individual doses 1000 m from the waste before 10,000 yr is less than the EPA limit line
     of 150 μSv.

2. Individual doses at the disposal boundary (5 km from waste disposal region) for a farm family using well water
from the underlying aquifer are very small.
   - Individual doses from the most important radionuclide, neptunium (\(^{237}\)Np)*, after 100,000 yr, at 1000 m
     from the waste, are 4.5 μSv for horizontal emplacement and 1 mSv for vertical emplacement; the doses
     were calculated using median parameter values.
   - Individual doses from neptunium at 1000 m from waste are 0.63 Sv when calculated using mean parameter
     values.

3. Individual dose from one occurrence in the first 10,000 yr of a driller closely examining a 100-mm-dia sample
from an MPC for one hour, and with continued incidental exposure to cuttings from the entire MPC for one
week thereafter (“groundshine”), is no more than 3 μSv.

4. The complementary cumulative distribution function of EPA summed normalized releases indicates the exam-
ined waste form (Section ES.2.2), container (Section ES.2.3), repository layout (Section ES.2.4), and geologic
barrier (Section ES.2.5) easily comply with the Containment Requirement of 40 CFR 191 (§13).

5. The radionuclides that contribute most to the EPA summed normalized release are as follows:* \(^{14}\)C, \(^{99}\)Tc, \(^{129}\)I,
\(^{234}\)U, and \(^{237}\)Np. (Because of the small inventory of \(^{14}\)C for the DOE spent fuel and its total absence in the
high-level waste, \(^{14}\)C did not cause a violation of the Containment Requirement of 40 CFR 191.)

6. For horizontal in-drift emplacement in an MPC, the probability of direct cuttings release is about eight times
greater than for vertical emplacement in boreholes; direct cuttings release is also about eight times greater for
horizontal emplacement than for vertical (assuming entire contents of container is brought to surface in each
case).

* The determination of which radionuclides are important contributors is also sensitive to the regulatory period being considered,
the performance metric (40 CFR 191 has been assumed here), and the location at which the metric is measured. This sensitiv-
ity and uncertainty regarding the form of the final regulations make it difficult, at this time, to identify radionuclides that should be
immobilized or removed during treatment.
Figure ES-2. Performance of tuff disposal system compared to various criteria. (a) Tuff disposal system complies with EPA 150-µSv dose criteria for first 10,000 yr; (b) doses of technetium-99 and neptunium-237 at well 1000 m downgradient of repository show neptunium-237 doses growing (deterministic simulation using median values); (c) tuff disposal system complies with EPA Containment Requirements; and (d) technetium-99 is main contributor to EPA summed normalized release at 2.4-km boundary in first 10,000 yr.
More important are results that can be used for project guidance with regard to a tuff disposal system. These are (1) the relative position of the various wastes in relation to (a) the DOE high-level waste in borosilicate glass and (b) the commercial pressurized water reactor (PWR) fuel, and (2) the relative importance of model parameters in explaining the value of a particular performance measure. These issues are explored in the following subsections.

**ES.2.2 Waste Form**

The waste form is the type of waste (e.g., graphite spent fuel) in combination with a specific treatment option. In the 1994 PA, only a minimal treatment option was considered, which means that the waste form and waste type are the same in most cases. The types of waste and treatment options are described below.

**Waste Types.** For the 1994 PA, seven waste types were examined, including five types of spent nuclear fuel (four from INEL and one from Hanford). The INEL high-level waste being modeled primarily originates with spent fuel that has been accepted for interim storage and reprocessing since 1953 by DOE at the Idaho Chemical Processing Plant. In addition, high-level waste from three other DOE sites was modeled. The seven waste types are listed in Table ES-2.

The waste types were studied individually in the Simple PA. When the waste types were studied as a group (Complex PA), radioactive waste produced and/or stored at the three DOE facilities was added to the hypothetical repository (1) to illustrate a more realistic mix of waste in the repository, (2) to simulate the impact of other DOE waste types on such a repository, and (3) as a reference for comparison with the other waste types, even though the waste from these DOE sites has been tentatively designated for Yucca Mountain, the proposed repository in Nevada for spent nuclear fuel from public utilities.

**Treatment.** Only the minimal waste treatment was considered in the 1994 PA for the spent fuel. This option essentially involves no conditioning, with one exception: the Advanced Test Reactor fuel, which represents the highly enriched spent fuel, was modeled as having been removed from the storage water basins, dried, and placed in borated stainless steel capsules before being inserted into the handling container. Some fuel elements of Advanced Test Reactor fuel have some type of damaged cladding—and thus an increased opportunity for creating critical conditions—and so placing this fuel in capsules is expected to facilitate packaging. However, in the 1994 PA, no protective capability was attributed to the stainless steel.

The N-Reactor fuel, which represents the weapons production fuel, also has damaged cladding, ranging from 16% to 50%, and removal from the reactor core. However, plans for dry storage for this fuel had not been defined at the time of this study, so no special treatment was assumed. Thus, radionuclides in the damaged N-Reactor fuel were available for immediate dissolution and/or diffusional release once the protective layers of the waste container failed.

Three treatment options were considered for the high-level waste stored at INEL (Table ES-3): (1) placement of the granular and powdered calcine loose into the container, (2) binding of the calcine in a glass ceramic using the Hot Isostatic Press (HIP) process, in which the calcine is mixed with glass additives, primarily silicon dioxide, heated up to 1323 K, and then compressed at 140 MPa as the mixture cools, and (3) vitrification of the calcine in borosilicate glass. For all three treatment options, the liquid high-level waste stored at INEL is first calcined. The first option is included for comparison with results from the 1993 PA. However, it does not meet the current 10 CFR 60 requirement, which prohibits particulate waste without consolidation (e.g., encapsulation in a matrix). Thus, the second treatment—glass ceramic—is the baseline.

Only one treatment option was considered for the high-level waste from Savannah River, Hanford, and West Valley: vitrification in borosilicate glass (Table ES-3).
### Table ES-2. Waste Types Studied in 1994 PA

<table>
<thead>
<tr>
<th>Waste Type</th>
<th>Specific Fuel Used to Represent Waste Type</th>
<th>Fuel Construction</th>
<th>MTHM*</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Spent Nuclear Fuel</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. Graphite</td>
<td>Fort St. Vrain, reactor in Colorado</td>
<td>Small particles of uranium and thorium carbide coated with silicon carbide in graphite matrix</td>
<td>28</td>
</tr>
<tr>
<td>2. Highly enriched uranium (HEU), i.e., fuel originally with over 20% by mass fissile 235U isotope</td>
<td>Advanced Test Reactor (ATR)</td>
<td>Uranium-aluminum alloy fuel clad by aluminum**</td>
<td>72</td>
</tr>
<tr>
<td>3. Highly enriched uranium in robust assemblies</td>
<td>Shippingport fuel from Light Water Breeder Reactor research program, Shippingport, Pennsylvania; inventory from Naval propulsion reactors included</td>
<td>Uranium dioxide fuel wafers and clad by zircaloy</td>
<td>110</td>
</tr>
<tr>
<td>4. Low enriched uranium (LEU) spent fuel, i.e., fuel originally with less than 20% by mass fissile 235U isotope</td>
<td>Commercial pressurized water reactor (PWR) fuel</td>
<td>Uranium-dioxide fuel pellets shaped into rods and clad by zircaloy</td>
<td>162</td>
</tr>
<tr>
<td>5. Weapons material fuel</td>
<td>N-Reactor fuel from Hanford, Washington; spent fuel from Savannah River not included</td>
<td>Uranium-metal fuel shaped into two concentric circles each clad with zircaloy</td>
<td>2100</td>
</tr>
<tr>
<td><strong>High-Level Waste</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>6. Calcine</td>
<td>Idaho Chemical Processing Plant at INEL</td>
<td>Calcination in a fluidized bed of liquid waste from reprocessing spent fuel to recover uranium; 3-18% powder, the remainder, granular</td>
<td>320***</td>
</tr>
<tr>
<td>7. Sludge Wastes</td>
<td>Savanah River (Defense Waste Processing Facility at Savannah River Plant in South Carolina)</td>
<td>Sludge liquids containing radionuclides from reprocessing spent fuel to recover uranium and plutonium</td>
<td>8036***</td>
</tr>
<tr>
<td></td>
<td>Hanford (Hanford Waste Vitrification Plant in Washington; double-shelled tanks only)</td>
<td></td>
<td>1160***</td>
</tr>
<tr>
<td></td>
<td>West Valley (West Valley Demonstration Project in New York)</td>
<td></td>
<td>72***</td>
</tr>
</tbody>
</table>

* Another initialism, MTIHM (metric tons of initial heavy metal), is used by some authors who wish to emphasize that the measurement is the initial mass of heavy metal rather than the current mass of heavy metal. In this report, we use the designation, MTHM, to mean initial mass, not only because it is found more frequently in the literature, but also because 40 CFR 191 defines "heavy metal" as "... all uranium, plutonium, or thorium placed into a reactor..." (40 CFR 191.12; emphasis added). Thus, the use of MTIHM, while useful in calling attention to this fact, is not necessary.

** Much of the foreign research reactor spent fuel being returned to the U.S. for disposal is aluminum clad.

***The equivalent mass of uranium, plutonium, and thorium represented by the high-level waste was calculated for all high-level waste as a group, using the most conservative procedure described in 40 CFR 191.
### Table ES-3. Treatment Options Used in the 1994 INEL Performance Assessment

<table>
<thead>
<tr>
<th>Category</th>
<th>Treatment Level</th>
<th>Description of Treatment Option</th>
</tr>
</thead>
<tbody>
<tr>
<td>Spent Nuclear Fuel</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Graphite (Fort St. Vrain)</td>
<td>Low</td>
<td>Placed directly in the waste parcel (whole block disposal)</td>
</tr>
<tr>
<td>Highly enriched uranium fuel (ATR)</td>
<td>Low</td>
<td>Dried, stabilized against further corrosion, and encapsulated in stainless steel before being placed in a handling container.</td>
</tr>
<tr>
<td>Highly enriched uranium fuel in robust assemblies (Shippingport)</td>
<td>Low</td>
<td>Placed directly in handling container; void space is not filled</td>
</tr>
<tr>
<td>Low enriched uranium (PWR)</td>
<td>Low</td>
<td>Placed directly in handling container; void space is not filled</td>
</tr>
<tr>
<td>Weapons production (N-Reactor)</td>
<td>Low</td>
<td>Dried and stabilized against further oxidation, placed in support structure in a handling container.</td>
</tr>
<tr>
<td>High-Level Waste*</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Calcine (INEL)</td>
<td>Low</td>
<td>Liquid high-level waste is calcined; this new waste and existing calcine waste are placed directly in containers.</td>
</tr>
<tr>
<td>Calcine (INEL)</td>
<td>Moderate</td>
<td>Liquid high-level waste is calcined and combined with existing calcine, glass frit, and immobilized as glass-ceramic in Hot Isostatic Press at 140 MPa and 1323 K.</td>
</tr>
<tr>
<td>Calcine (INEL) and DOE sludge wastes (Savannah River, Hanford, and West Valley)</td>
<td>Moderate</td>
<td>Immobilized in borosilicate glass.</td>
</tr>
</tbody>
</table>

* Constraints on the waste treatment options for calcine, such as required development time or costs, have not yet been evaluated for their conformance with desired timetables or economic realism.

The waste types and treatments are depicted in Figures ES-3 and ES-4. The parameters for modeling the waste form that were varied, and thus contribute to the uncertainty in the results, are shown in Table ES-4.

Table ES-5 is a summary of the information acquired about the waste form. Figure ES-5 shows the behavior of individual waste forms.
Figure ES-3. Packaging for horizontal emplacement in unsaturated tuff. Schematic depicts placement of most waste forms directly into Multi-Purpose Canisters (MPC), which corresponds to minimal treatment and which was examined for the 1994 performance assessment of spent fuel and high-level waste owned by DOE. The MPC handling containers are overpacked with two disposal containers: an inner container of 20-mm Inconel 825 and an outer container of 100-mm carbon steel.
Figure ES-4. Packaging for vertical emplacement in unsaturated tuff. Schematic depicts placement of most waste forms directly into Low Weight Track (LWT) canisters, which corresponds to minimal treatment and which was examined for the 1994 performance assessment of spent fuel and high-level waste owned by DOE. The LWT canisters and the canister container are shown overpacked with two disposal containers: an inner container of Inconel 825 and an outer container of thick carbon steel. Vertical emplacement is not currently contemplated for the potential repository at Yucca Mountain but was included for comparison with the results of the 1993 PA.
**Table ES-4. Parameters Varied for Waste Form**

<table>
<thead>
<tr>
<th>Number</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.*</td>
<td>Fraction of graphite fuel represented by Fort St. Vrain with initially failed silicon carbide coatings.</td>
</tr>
<tr>
<td>2.</td>
<td>Fraction of fuel from N-Reactor with initially failed zircaloy cladding.</td>
</tr>
<tr>
<td>3.</td>
<td>Fraction of robust, highly enriched uranium fuel represented by the Shippingport fuel with initially failed zircaloy cladding.</td>
</tr>
<tr>
<td>4.</td>
<td>Alteration rate of ceramic phase in glass-ceramic matrix immobilizing calcine high-level waste.</td>
</tr>
<tr>
<td>5.</td>
<td>Alteration rate of glass immobilizing high-level waste from all sources.</td>
</tr>
<tr>
<td>6.</td>
<td>Alteration rate of uranium dioxide fuel in robust highly enriched fuel represented by Shippingport fuel and low enriched fuel represented by pressurized water reactor fuel (PWR).</td>
</tr>
<tr>
<td>7.</td>
<td>Solubility of neptunium (Np) in water not saturated with silica.</td>
</tr>
<tr>
<td>8.</td>
<td>Solubility of protactinium (Pa), assumed for water both saturated and not saturated with silica.</td>
</tr>
<tr>
<td>9.</td>
<td>Solubility of plutonium (Pu) in water not saturated with silica; also assigned to americium (Am) and samarium (Sm).</td>
</tr>
<tr>
<td>10.</td>
<td>Solubility of uranium (U) in water not saturated with silica.</td>
</tr>
<tr>
<td>11.</td>
<td>Solubility of neptunium (Np) in water saturated with silica.</td>
</tr>
<tr>
<td>12.</td>
<td>Solubility of plutonium (Pu) in water saturated with silica; also assigned to americium (Am) and samarium (Sm).</td>
</tr>
<tr>
<td>13.</td>
<td>Solubility of uranium (U) in water saturated with silica.</td>
</tr>
<tr>
<td>14.</td>
<td>Fraction of inventory of carbon-14 ($^{14}$C) that is present in the gap between the zircaloy cladding and the uranium dioxide ($^{233}$UO$_2$) or in grain boundaries of the Shippingport fuel or pressurized water reactor (PWR) fuel where it may be released immediately when the cladding fails.</td>
</tr>
<tr>
<td>15.</td>
<td>Fraction of inventory of cesium-137 ($^{137}$Cs) that is present in gap and grain boundaries of the UO$_2$ matrix of the Shippingport or PWR fuel.</td>
</tr>
<tr>
<td>16.</td>
<td>Fraction of inventory of iodine-129 ($^{129}$I) that is present in gap and grain boundaries of the UO$_2$ matrix of the Shippingport or PWR fuel.</td>
</tr>
<tr>
<td>17.</td>
<td>Fraction of inventory of technetium-99 ($^{99}$Tc) that is present in gap and grain boundaries of the UO$_2$ matrix of the Shippingport or PWR fuel.</td>
</tr>
</tbody>
</table>

* Numbers correspond to parameter numbering scheme in Volume 2, Chapter 8, Table 8-2.
1. Each waste form complies with Containment Requirements of 40 CFR 191 when placed individually in a repository; some even comply at the boundary of the waste parcel.

2. Difference in performance of the waste types were due primarily to the temperature-sensitive corrosion rates and thus were influenced by differences in local temperatures around the container, which fluctuated because of substantial differences in heat output of the waste.
   - Other differences in waste types that affect performance include type and integrity of cladding, corrosion rate of form, and inventory (e.g., high-level waste does not contain $^{14}$C).

3. In general, high-level waste performs better than spent nuclear fuel.

4. The Fort St. Vrain graphite fuel and INEL calcine high-level waste were too cool to obtain a 23.5 W/m$^2$ power density when placed individually in the repository (see Section ES.2.4, Repository Design); hence, the INconel Alloy 825 layer of the disposal container (see Section ES.2.3, Waste Container) did not fail for these waste types.
   - For simulations where the INconel Alloy 825 layer is prebreached, the INEL calcine waste performs about the same as the DOE high-level waste within a repository as measured by the EPA summed normalized releases.
   - The rank order from best to worst is:
     1. Calcine immobilized in glass-ceramic
     2. Calcine immobilized in borosilicate glass
     3. Savannah River sludge waste immobilized in borosilicate glass
     4. Hanford sludge waste immobilized in borosilicate glass
     5. West Valley sludge waste immobilized in borosilicate glass
     6. Calcine loose (similar to N-Reactor fuel and still within compliance; this latter option is not currently allowed because of prohibitions in 10 CFR 60 concerning particulates).
   - Calcine performs much better than the DOE glass in the geologic barrier because of the low inventory of technetium-99; this low inventory is possibly artificial and needs to be verified in future work.

5. The performance of DOE spent fuels straddles the performance of pressurized water reactor (PWR) spent fuels when the measure is the EPA summed normalized release.
   - The rank order from best to worst is (order controlled by integrity or durability of cladding):
     1. Fort St. Vrain graphite fuels (uranium and thorium carbide with silicon carbide coating) (similar to N-Reactor fuel when there is no INconel Alloy 825 layer in disposal container)
     2. Shippingport (uranium dioxide fuel with zircaloy cladding)
     3. PWR (uranium dioxide fuel with zircaloy cladding)
     4. N-Reactor (uranium metal fuel with damaged zircaloy cladding)
     5. ATR fuel (uranium metal fuel with aluminum cladding)

6. The heat of reaction of oxidizing all the uranium metal in N-Reactor fuel is about $1.34 \times 10^{13}$ J or one-tenth of the heat energy produced in one year by radioactive decay of a 12,060-MTHM repository. Hence, while not small, the energy is not significant on a global scale in the 1994 PA calculation.
Behavior of waste forms when placed individually in tuff disposal system. All waste forms in disposal containers comply with EPA Containment Requirements in first 10,000 yr (Simple PA). Hence, releases at the waste parcel and water table are shown on the left. Because temperature around the container greatly influences performance, the total heat output from the repository for all wastes and the power density of all wastes, except the calcine high-level waste and the Fort St. Vrain graphite fuel, were kept the same in each configuration. This method of comparison meant that the physical size of the repository and the quantity of waste (MTHM) varied per configuration. The calcine and the Fort St. Vrain fuel were cool enough to produce slow corrosion rates, which meant that the Inconel Alloy 825 layer did not fail in 10,000 yr; thus releases without the Incoloy Alloy 825 layer are shown on the right.
ES.2.3 Waste Container

In the 1994 PA, waste containers include both the handling container and up to two disposal containers (overpacks). In general, three types of waste handling containers were modeled in the 1994 PA: the 125-ton Multi-Purpose Canister (MPC) for horizontal emplacement, the 25-ton Legal Weight Truck (LWT) canister for vertical emplacement, and the DOE standard high-level waste container (used for both vertical and horizontal emplacement). (See Section ES.2.4 for a discussion of emplacement schemes.) The material for all three handling containers is stainless steel type 304L. The thickness of the MPC is 25.4 mm; both the LWT container and the DOE standard container are 9.5 mm. None of the handling containers uses a backfill (filler material).

The waste canisters are placed in two disposal containers (overpacks), one of 100-mm-thick corrosion-allowance material (i.e., carbon steel) and the other of a 20-mm-thick corrosion-resistant material (Incoloy 825) as an additional inner barrier (Figures ES-3 and ES-4). However, one series of calculations assumed the Incoloy 825 layer was present but breached; the purpose was to evaluate the effects of this container layer. The MPC that was modeled in the 1994 PA is based on the storage and transport container currently considered by the Yucca Mountain Project. The MPC is expected to be transported on a rail system; the LWT could be transported by truck. Specifics about the LWT are not currently defined since the Yucca Mountain Project is no longer actively considering vertical emplacement.

Spent Fuel. Spent fuel with minimal conditioning is assumed to be placed directly into the 125-ton MPCs or the 25-ton LWT.

High-Level Waste. For horizontal emplacement, the 1994 PA assumed the small handling containers of both INEL and DOE high-level waste will be placed inside MPCs and then into disposal containers (overpacks). For vertical emplacement, handling containers of calcine high-level waste from INEL are placed in disposal containers of corrosion-resistant material, and then into boreholes.

Because of the diverse types of waste under study and the need for interaction with the flow and transport codes, the 1994 PA continued development of the source term submodel to include important processes (Figure ES-6). The capabilities of the submodel are fully described in Volume 2, Chapter 11; an indication of these capabilities can be seen from the list of parameters that were varied for the waste form (Table ES-4) and the waste container (Table ES-6). Table ES-7 summarizes the information gathered about the waste containers. Figure ES-7 shows performance of waste containers compared to current waste package and engineered barrier criteria in 10 CFR 60.
For Dripping to Occur, Model Assumes Fractures Must be Saturated

For Water from Porous Matrix to Contribute to Corrosion, Model Assumes Container must Contact Tuff (Rubble or Collapse of Container on Floor)

Radionuclides in Gap between Pellet and Cladding or Grain Boundaries (Released Instantly to Pore Fluid when Exposed)

Radionuclides in Matrix (Radionuclides Released Congruently with Matrix Alteration which is Controlled by Arrhenius Rate or Solubility)

Radionuclides Precipitates (Unbound Solids) within Canister

Radionuclides in Pore Fluid; Released by Diffusion (I-D) Across Rubble or to any Adverting Fluid (up to Solubility Limit)

Radionuclide Sorption on Corrosion Products of the Container

Radionuclides in Matrix (Radionuclides Released Congruently with Matrix Alteration which is Controlled by Arrhenius Rate or Solubility)

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Figure ES-6. Locations of radionuclides in waste container. The source term submodel monitors these locations as protective layers are breached. The model must accommodate 9 sources of waste, each with a different radionuclide inventory, up to 8 nested layers that may react with water, and 10 different matrices, such as uranium dioxide and borosilicate glass.

Table ES-6. Parameters Varied for the Waste Container

18. Alteration rate of 304L stainless steel. Stainless steel is found in the handling container, borated stainless steel support structures, and stainless steel capsules. The alteration rate affects the release rates of the boron neutron sorber, but does not affect the breach times because no protective function was attributed to the stainless steel because of the high probability of stress cracking when temperatures are above 60°C. This sampled value was used for both fully saturated and unsaturated (i.e., moist) steel.

19. Alteration rate of Inconel 825 disposal container.

20. Alteration rate of ASTM Type A-516 Grade 60 carbon steel disposal container.

21. Effective area and tortuosity term for diffusion of radionuclides through the breached container.

22. Adsorption coefficient of plutonium on rust of outer carbon steel disposal container.

23. Adsorption coefficient of uranium on rust of outer carbon steel disposal container; also assigned to selenium.

24. Area above the container that contributes water (e.g., drips), which in turn, contacts the container surface.

25. Fraction of carbon steel disposal containers breached when placed into the repository and thus does not protect the underlying Inconel 825 layer.

* Numbers correspond to parameter numbering scheme in Volume 2, Chapter 8, Table 8-2.
Table ES-7. Summary of Information Gathered about Waste Containers

1. Based on current literature values for corrosion rates of carbon steel and Inconel Alloy 825 at temperatures near 100°C with oxygen and water*, many of the disposal containers breach rapidly. However, breaching of the disposal containers does not necessarily imply complete release of radionuclides because other protective layers must also breach and the matrix containing the radionuclides be altered.
   - The mean** fraction of waste form exposed to groundwater (weighted by the MTHM of each waste form) is more than 10% at 300 yr. (About 60% of the simulations showed between $10^{-5}$ and $7 \times 10^{-4}$ of the waste exposed in the first 300 yr).
   - However, the fraction of waste altered (weighted by the MTHM of each waste form) and ready for transport is only about 1% at 300 yr.
   - Fraction of waste exposed is much greater than the fraction altered during the first 10,000 yr due to the slow dissolution rate of glass matrices.
   - Jumps in the fraction of waste altered are due to exposure of fast reacting matrices such as in the ATR and N-Reactor fuels.

2. Several of the radionuclide release rates exceed the allowable rate from the engineered barrier specified in 10 CFR 60.
   - The mean release rates of $^{99m}$Tc and $^{14}$C exceed the allowable release rates.
   - Only the 90% quantiles of $^{129}$I and $^{238}$U exceed the allowable release rates.

3. The retardation of radionuclides by sorption onto rust particles (assuming adequate oxygen) only somewhat reduced releases of liquid-transported radionuclides to the host rock.

4. The high-level waste performs well enough that a robust disposal container is not needed except for compliance with 10 CFR 60.

* The reader should consult Volume 2, Chapter 11, for a full description of the temperature-dependent rates.
** The NRC does not state in 10 CFR 60 precisely what "substantially complete" means in quantitative terms nor what probability should be used to compare release rates.
Figure ES-7. Performance of waste package and engineered barrier for (a) fraction of waste exposed between 0-300 yr and 0-100,000 yr (median), and (b) release rate of technetium-99.
ES.2.4 Repository Design

The repository is the underground portion of a waste disposal facility, including the disposal area, access drifts, ramps, and disturbed rock around these features. Neither the above-ground facilities nor the undisturbed rock is included in our definition of the repository. The repository may include ventilation shafts and underground operations areas, but none was assumed for this generic tuff disposal system.

The disposal region of the repository was sized to accommodate 12,060 MTHM of spent fuel or equivalent high-level waste. No other waste was included in the repository. The design included long waste disposal rooms (tunnels), surrounded on the perimeter by an access drift, which in turn leads to ramps that connect to the surface (Figure ES-8). The disposal region was assumed to be excavated by a tunnel boring machine.

Two orientations for waste emplacement were considered: vertical emplacement in boreholes in the floors of the waste disposal rooms (similar to that considered in the 1993 PA), and horizontal emplacement, in which the containers are transported to the rooms by rail and then placed end-to-end. The vertical emplacement scheme allowed comparisons between the tuff repository and the salt and granite repositories considered in the 1993 PA; vertical emplacement was also the original concept for proposed waste disposal at Yucca Mountain although it is no longer considered. The horizontal emplacement scheme is the current concept for waste disposal at the proposed repository at Yucca Mountain. No backfill is used directly around the waste container in either orientation.

Figure ES-8. Layout of waste disposal rooms and access drifts in tuff for hypothetical repository containing DOE spent fuel and high-level waste.
For vertical emplacement, the rooms were spaced 7.5 m apart and the boreholes in the floors of the rooms were spaced 2.6 m apart, which corresponds to an areal design power density of about 14.1 W/m² for the mixture of waste in the Complex PA. For horizontal emplacement, the rooms were spaced 4.3 m apart (as close as thought possible to maintain structural integrity) and the waste containers were placed end to end, which corresponds to an areal power density of about 23.5 W/m² for the mixture of waste in the Complex PA.

The parameter that was varied for modeling the repository was as follows (number corresponds to parameter listed in Volume 2, Chapter 8, Table 8-2):

26. Fraction of waste containers that has direct contact with water in the matrix of the host rock for a significant portion of the 10,000-yr regulatory period (value is constant with no time dependence); contact is envisioned to occur when either the air gap between the container and the host rock fills with rubble, or the rail car collapses from corrosion and allows the disposal container to contact an undrained room floor.

Table ES-8 summarizes the information acquired about the repository in the tuff disposal system. Figure ES-9 provides results of flow behavior.

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Table ES-8. Summary of Information Gathered about the Repository from Complex PA

1. The thermal power density and its influence on local temperatures in the repository has a very strong effect on the behavior of liquid and gas flow through and around the repository (Figure ES-9).

2. The 65% water saturation in the horizontal emplacement scheme in the tuff matrix is sufficient to keep temperatures in computational grid blocks below 100°C as water vaporizes. Thus maximum local temperatures in the repository are substantially lower than those calculated when only heat conduction and radioactive heat transfer are considered.

   - In the horizontal emplacement scheme, repository temperatures increase from an initial value of 26°C to a maximum value of about 93°C (thus, no dry-out zone), and then fall slowly. The peak temperature is reached at the center of the repository, about 30 yr after waste disposal.

   - In the vertical emplacement scheme, the maximum repository temperature is about 25°C lower than for horizontal emplacement.

   - When only heat conduction and radioactive heat transfer are considered, the maximum repository temperature for horizontal emplacement is about 134°C.

3. Between 0 and 500 yr, gas flow is dominated by vaporization of water in the repository, movement out in all directions in the highly conductive fracture network, and then condensation all around the repository.

   - Because of water vaporization, the air mass percentage drops from an initial value of 92% to 30%. Assuming the mole percentage of oxygen in air is 21%, the oxygen percentage in the gas phase drops to 7% by mass and 4.4% by moles. Because anoxic corrosion rates are generally lower than oxic corrosion rates, the lack of oxygen could counteract somewhat the greatly increased corrosion rate from the temperature increase during the first 500 yr (period dominated by local gas movement).

   - This vaporization process produces a rise in gas pressure. In the horizontal and vertical emplacement schemes, the pressure rises to a maximum of 0.1044 and 0.1042 MPa, respectively. The corresponding drop in liquid pressure draws water towards the repository. In the horizontal emplacement scheme, water is drawn from as far away as the water table, located 300 m below the repository.

4. After 1000 yr, gas flow is dominated by the regional circulation of gas caused from the heating of the area around the repository; the regional circulation does not diminish substantially over the next 9000 yr.

   - General heat conduction through the rock matrix adequately predicts the general temperature profile on a regional scale after 1000 yr.

   - The travel time to the surface of gas originating in the repository is fastest once the regional gas flow circulation pattern has been established.

5. Gas consumption from oxic corrosion of the containers had little influence on the flow of gas.
Figure ES-9. Fluid and gas flow behavior. Fluid flow behavior prior to 500 yr is dominated by the vaporization and condensation of water around the repository, while fluid flow behavior after 1000 yr is dominated by the regional circulation of gas from the general temperature rise around the repository.
ES.2.5 Geologic Barrier

The geologic barrier of the tuff disposal system comprises the sequence of tuff that isolates the repository from the accessible environment. The accessible environment is defined by 40 CFR 191 as the surface and any location elsewhere that is beyond 5 km from the repository. For consistency with the assessment of the granite and salt disposal systems, a boundary 2.4 km from the edge of the disposal region was used in the 1994 PA.

The stratigraphy of the tuff disposal system is idealized as a series of constant thickness hydrologic modeling units ("pancakes") with a dip of 4.6°. The units consist of consecutive layers of tuff with degrees of welding and porosity that are similar (Figure ES-10). The repository is assumed to be located in the unsaturated zone about 300 m below the surface and 300 m above an aquifer.

In the unsaturated zone, the composite porosity simplification (also called equivalent continuum) of the dual permeability model is used for fluid flow, i.e., the fractures and matrix are assumed to be in local thermodynamic equilibrium. Thus a single saturation and permeability function is used to represent the fracture and matrix media. For transport in the unsaturated zone, the model uses a full dual-permeability formulation, and so fluid velocities for fracture and matrix must be estimated from the composite velocities. One phase (liquid or gas) is tracked at a time. In the saturated zone, the dual porosity simplification of the dual permeability model is used for both liquid flow and transport, i.e., the permeability (but not porosity) of the matrix is assumed to be zero.

For the Complex PA, two-phase flow (liquid and gas) with heat conduction, convection, and phase changes are included in the two-dimensional model of the unsaturated zone. The cross-section modeled is perpendicular to the 4.6° dip. The 1° dip was assumed minor in this direction and so the model uses horizontal layering.

Because the emphasis of this performance assessment is on the behavior of the waste form and container, many parameters for the geologic site were not varied. Also, the interpreted data from past assessments of the proposed repository at Yucca Mountain by Sandia were used for all geologic components of the disposal system (such as permeability of the unsaturated tuff) without reevaluation. The parameters that were varied for modeling the repository are shown in Table ES-9. Table ES-10 summarizes the information gathered about the geologic barrier. Figure ES-11 shows the effect of the geologic barrier.
Figure ES-10. Correspondence of stratigraphic column of Yucca Mountain tuffs and hydrologic modeling units used in 1994 PA. For the Simple PA, the units were modeled in one dimension. For the Complex PA, the unsaturated zone was modeled in two dimensions, perpendicular to the 4.6° dip (see Figure ES-4), and the saturated zone in three dimensions, as depicted.
Table ES-9. Parameters Varied for Geologic Barrier

| 27. | Bulk, saturated hydraulic conductivity that was assigned to TSw, TSv, and TCw modeling layers (portion of Topopah Spring consisting of welded tuff and vitric tuff, and portion of Tiva Canyon tuff). |
| 28. | Bulk, saturated hydraulic conductivity that was assigned to CHnv and CHnz modeling layers (portion of Calico Hills consisting of nonwelded, vitric and nonwelded, zeolitic tuff). |
| 29. | Bulk hydraulic conductivity of PPw modeling layers (portion of Prow Pass tuff). |
| 30. | Hydraulic conductivity of matrix material that was assigned to TSw, TCw, and CHnz modeling layers (portion of Topopah Spring consisting of welded tuff, portion of Tiva Canyon tuff, and portion of Calico Hills consisting of nonwelded, zeolitic tuff). |
| 31. | Hydraulic conductivity of matrix material that was assigned to CHnv and PTn modeling layers (portion of Calico Hills consisting of nonwelded, vitric and Paint Brush tuff). |
| 32. | Hydraulic conductivity of matrix PPw modeling layer. |
| 33. | Coefficient $\beta$ for Van Genuchten capillary pressure model (model of two-phase saturation) that was assigned to TSw, TCw, and CHnz modeling layers. |
| 34. | Coefficient $\beta$ for Van Genuchten capillary pressure model that was assigned to CHnv and PTn modeling layers. |
| 35. | Coefficient $\beta$ for Van Genuchten model of two-phase saturation that was assigned to PPw modeling layer (portion of Prow Pass tuff). |
| 36. | Spacing of fractures that was assigned to TSw, TSv, and TCw modeling layers. |
| 37. | Spacing of fractures that was assigned to CHnv, CHnz, and PTn modeling layers. |
| 38. | Spacing of fractures in PPw modeling layer. |
| 39. | Adsorption coefficient of neptunium (Np) in devitrified tuff (TCw, TSw, PPw, BFw, and TRw modeling layers); full value was assigned to protactinium (Pa); one-quarter of the sampled value was assigned to neptunium and protactinium in vitric tuffs (PTn and CHnv modeling layers). |
| 40. | Adsorption coefficient of plutonium (Pu) in vitric and devitrified tuff (TCw, PTn, TSw, CHnv, PPw, BFw, and TRw modeling layers). |
| 41. | Adsorption coefficient of uranium (U) in vitric and devitrified tuff (TCw, PTn, TSw, CHnv, PPw, BFw, and TRw modeling layers); value also assigned to selenium (Se). |
| 42. | Adsorption coefficient of neptunium (Np) in zeolitic tuff (CHnz, CFUn, and CHMn modeling layers); value also assigned to protactinium (Pa). |
| 43. | Adsorption coefficient of plutonium (Pu) in zeolitic tuff (CHnz, CFUn, and CHMn modeling layers). |
| 44. | Adsorption coefficient of uranium (U) in zeolitic tuff (CHnz, CFUn, and CHMn modeling layers); also assigned to selenium (Se). |
| 45. | Fraction of contaminant travel distance set equivalent to longitudinal dispersivity ($\alpha_L$). |

* Numbers correspond to parameter numbering scheme in Volume 2, Chapter 8, Table 8-2.
Inclusion of sorption of radionuclides on tuff within the unsaturated and saturated zones greatly diminished the release of radionuclides. The only radionuclides to reach the water table in 10,000 yr were minute amounts of uranium and neptunium, because of their high initial quantity and only moderate sorption, and larger amounts of radionuclides that are not readily sorbed. No radionuclides in meaningful quantities entering the saturated zone reached the 2.4-km boundary in 10,000 yr.

The reduced solubility assumed for uranium in water saturated with silica and the moderate sorption assumed greatly diminished the amount of uranium reaching the water table. In addition, sorption of plutonium by tuff inhibited its release from the waste disposal region and prevented plutonium from even reaching the water table.

Between 10,000 and 50,000 yr, additional amounts of uranium and large amounts of neptunium reach the water table.

By 100,000 yr, results from the deterministic run using mean values show that about 0.156 kg each of $^{237}$Np and $^{129}$I, and 3.37 kg of $^{99}$Tc (about 57 curies total), have left the 2.4-km boundary.

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Table ES-10. Summary of Information Gathered about the Geologic Barrier

<table>
<thead>
<tr>
<th>1</th>
<th>Inclusion of sorption of radionuclides on tuff within the unsaturated and saturated zones greatly diminished the release of radionuclides.</th>
</tr>
</thead>
<tbody>
<tr>
<td>2</td>
<td>The only radionuclides to reach the water table in 10,000 yr were minute amounts of uranium and neptunium, because of their high initial quantity and only moderate sorption, and larger amounts of radionuclides that are not readily sorbed ($^{129}$I, and $^{99}$Tc).</td>
</tr>
<tr>
<td>3</td>
<td>No radionuclides in meaningful quantities entering the saturated zone reached the 2.4-km boundary in 10,000 yr.</td>
</tr>
<tr>
<td>4</td>
<td>The reduced solubility assumed for uranium in water saturated with silica and the moderate sorption assumed greatly diminished the amount of uranium reaching the water table. In addition, sorption of plutonium by tuff inhibited its release from the waste disposal region and prevented plutonium from even reaching the water table.</td>
</tr>
<tr>
<td>5</td>
<td>Between 10,000 and 50,000 yr, additional amounts of uranium ($^{233}$U, $^{234}$U) and large amounts of neptunium ($^{233}$Np) reach the water table.</td>
</tr>
<tr>
<td>6</td>
<td>By 100,000 yr, results from the deterministic run using mean values show that about 0.156 kg each of $^{237}$Np and $^{129}$I, and 3.37 kg of $^{99}$Tc (about 57 curies total), have left the 2.4-km boundary.</td>
</tr>
</tbody>
</table>
Figure ES-11. Effect of geologic barrier. Total cumulative releases, cumulative releases of individual radionuclides, and concentrations from deterministic simulation to 100,000 yr using mean values of all model parameters. Results show the unsaturated zone greatly reduces releases in the first 30,000 yr and the saturated zone greatly reduces releases for at least 100,000 yr.
ES.2.6 Agents Acting on the Tuff Disposal System

The agents acting on the tuff disposal system that were considered in the 1994 PA included human intrusion from exploratory drilling and cyclical climatic change.

For human intrusion, exploration for natural resources (both economic minerals and hydrocarbons) was assumed to occur using present methods of drilling. Thus, the size of drill bits ranged from 5 cm (2 in.) to 61 cm (2 ft) in diameter. The drilling rate was calculated assuming a random Poisson process, with the maximum drilling rate of 3 boreholes per km$^2$ (the same rate was used for the granite site in the 1993 PA). Because tuff is already fractured, the drilling was assumed to provide only an avenue for direct release of radionuclides, not a substantial increase in permeability leading to increased liquid flow. This same assumption was used for the granite site (1993 PA) and the assessment of the potential repository at Yucca Mountain (TSPA-1993, Wilson et al., 1994), but has not been evaluated.

Climatic change was included because it could enhance infiltration and thus percolation of groundwater through the unsaturated zone. Earlier performance assessments of the proposed repository at Yucca Mountain also included climatic change. In the 1994 PA, a sinusoidal model was used to model the potential change in infiltration, and was intended to predict the time when changes occurred. The mean for the distribution of infiltration for dry climates was 0.5 mm/yr; the mean infiltration for wet climates was 10 mm/yr. To account for a potentially large ponding of water at the surface and flow down connected fractures, part of this infiltration was concentrated in one region of the two-dimensional model for the Complex PA. In the Simple PA, this feature was unnecessary. Because the flow field in the Simple PA's one-dimensional consequence model was evaluated assuming steady state and sampled climatic changes were generally small in the first 10,000 yr, the infiltration was taken as average infiltration over 10,000 yr.

The parameters that were varied when modeling the geologic barrier are shown in Table ES-11. Table ES-12 summarizes the information gathered about the agents acting on the tuff disposal system, and Figure ES-12 provides support for these findings.

<table>
<thead>
<tr>
<th>Number</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>46.*</td>
<td>Infiltration rate above repository for current dry climate.</td>
</tr>
<tr>
<td>47.</td>
<td>Maximum increase in infiltration above repository during wet climate.</td>
</tr>
<tr>
<td>48.</td>
<td>Period of climate change.</td>
</tr>
<tr>
<td>49.</td>
<td>Potential concentration of infiltration because of surface or subsurface features above the repository.</td>
</tr>
<tr>
<td>50.</td>
<td>Diameter of intrusion borehole.</td>
</tr>
<tr>
<td>51.</td>
<td>Frequency of intrusion into the repository based on Poisson analytic function.</td>
</tr>
</tbody>
</table>

* Numbers correspond to parameter numbering scheme in Volume 2, Chapter 8, Table 8-2.
Table ES-12. Summary of Information Gathered about the Agents Acting on the Tuff Disposal System

1. Climatic change and its assumed direct influence on water infiltration had only a minor influence flow of water to water table in the 10,000-yr simulation* in the Complex PA using a two-dimensional model.
   - For horizontal emplacement, many simulations showed cumulative flow from the water table to the repository as the result of lower liquid pressures as water is vaporized around the repository in the first 1000 yr.
   - All particles released at the surface to track percolating water were trapped in the high porosity PTn layer above the repository, which suggests that little flow from the surface passes into the TSw layer until the saturation level of the matrix is raised enough to allow some flow in the fractures.

2. Potential concentration of infiltration or percolation above the repository is influential in getting water to the repository after about 3000 yr by changing the flow regime from matrix flow to fracture flow in the Complex PA.

3. In contrast to the Complex PA, the three parameters for climatic change (see Table ES-11) were the most important model parameters in the Simple PA.

* 40 CFR 191 clearly stated in the preamble to the 1985 promulgation of the regulation that a time period was purposely selected by the EPA such that changes occurring on a geologic time scale would not affect the compliance evaluation.
Executive Summary

Climatic Changes are not Dramatic in the First 10,000 yr

Figure ES-12. Results of agents acting on disposal system. In contrast to the Simple PA, change in water infiltration from climatic variations had much less influence on water flow to the water table until the end of the 10,000-yr simulation in the Complex PA. In fact, water flows from the water table in the first 3000 yr in several simulations and very little infiltrating water makes it to the water table.
ES.2.7 Critical Conditions in or near Containers

The 1994 PA did not limit the amount of fissile material (e.g., \(^{233}\text{U}, ^{234}\text{U}, ^{234}\text{Pu}, ^{239}\text{Pu}\)) in a container as was the case in the 1993 PA; instead, the 1994 PA modeled as many fuel assemblies as would reasonably fit into a container (in this case, an MPC), which corresponds to up to 1216 kg of fissile mass (926 kg of \(^{235}\text{U}\)) per large multi-purpose canister if 33 containers are used (Section ES.2.2 and Figure ES-3). Therefore the 1994 PA studied the possibility of conditions occurring that were conducive to a criticality in or near a container during the 10,000-yr regulatory period or beyond.

Two important aspects of the criticality to consider are (1) the assembly of the fissile material into a configuration that promotes a nuclear chain reaction and (2) the continued operation of the nuclear chain reaction within the fissile mass. Within the time available for the 1994 PA, neither aspect could be explored completely but both were studied. The following text first discusses the operation of the assembly and then how fissile material might be assembled for a nuclear reaction. It is important to note, however, that these two aspects are interrelated because the processes that assemble the fissile mass establish the initial conditions for the operation of the nuclear reaction. For a credible analysis, it is important to maintain consistency between these two aspects.

Bounding Consequences of a Critical Condition. One approach to examining a criticality in a repository is simply to assume a critical condition and examine the consequences. If the consequences are negligible, then a basis is established for ignoring the scenario. The disadvantage of this approach is that another analysis with different, and improbable, initial conditions might generate results that incorrectly indicate large nuclear excursions. Thus, it is imperative that the analysis consider how fissile mass might assemble in a repository (discussed below) and also maintain consistency with that process. The bounding analysis performed for the 1994 PA, consistent with the manner in which fissile mass is envisioned to assemble, showed that the consequences of a criticality would be minimal. These included (a) producing more radionuclides from the fission process, (b) generating fission gases, (c) generating additional heat, (d) physically damaging the tuff around the container through a rapid pressure rise as steam is produced, and (e) creating a criticality at the surface during exploratory drilling. The consequences of physical damage to the tuff were not studied in the 1994 PA because it was considered less likely than the other scenarios, based on the evaluation of required processes described below, and it would also require a new modeling capability. The criticality at the surface was omitted in the 1994 PA because it already represented a release and required assumptions about details of drilling practices that were beyond the scope of this year's work.

Relevant natural analogues, accidents, and experimental criticalities can be used as one method to bound the consequences of the critical condition. For this purpose, the relevant natural analogues, accidents, and experimental criticalities can be categorized to form an event tree according to the degree of moderation and the rate of fissile material assembly: (1) moderated, low rate of assembly, typified by the natural Oklo reactors in Africa, (2) unmoderated, low rate of assembly, typified by a fast breeder reactor, (3) moderated, high rate of assembly, typified by accidents with aqueous solutions in reprocessing plants, and (4) unmoderated, high rate of assembly, typified by experiments and accidents with critical assemblies.

Criticality types 3 and 4 (moderated and unmoderated with high rate of assembly) were characterized by power pulses ranging from \(10^{17}\) to \(10^{18}\) fissions (nuclear excursion involving prompt neutron fissions). Bounding calculations revealed that a local critical event of types 1 and 2 (moderated and unmoderated with a low rate of assembly) which are bounded by a steady-state fission process that maintains temperatures below boiling for water in the container, produces no more than \(10^{25}\) fissions during 10,000 yr. Assuming a 12,060-MTHM repository of spent fuel with an average burnup of between 25,000 and 40,000 MWd/MTHM, the number of fissions that is represented by the spent fuel is on the order of \(10^{30}\) fissions. Consequently, an increase in the inventory of fission products from one critical event of \(10^{18}\) fissions is negligible; only those radionuclides with very short half-lives have an appreciable increase in comparison to the original inventory. Similarly, the impact of \(10^{25}\) atoms split from a critical condition occurring in a container in the repository, assuming a steady-state reactor, is still negligible. Furthermore, if one assumes that 289 MPCs with highly enriched uranium spent fuel all go critical, about \(10^{28}\) fissions occur, which is only 1% of the inventory when represented as fissions for a 12,060-MTHM repository with 25,000 to 40,000 MWd/MTHM burnup. Thus, no important consequences were observed.

**** The burnup of the DOE spent fuel is typically much lower than this value, as explained in Volume 3, Appendix A. The average burnup is used as a conservative value for the bounding calculations.
Assembly of Fissile Material. The containers under study are designed to preclude criticalities even when fully filled with water by incorporating neutron absorbers, such as a borated stainless steel support structure for the spent nuclear fuel. After disposal, neutron interaction between containers is avoided for wet or dry containers either by ensuring sufficient spacing between containers (2.6-m spacing for vertical emplacement) and/or sufficient container material (stainless steel, Incoloy 825, and carbon steel for both the LWT and MPC containers) (See also the criticality modeling of containers in Volume 2, Chapter 10, and Volume 3, Appendix E.)

Given these design assumptions, a critical condition can occur after disposal only if, first, the container (and any other protective layer such as cladding) corrodes and, then, either the fissile material goes into solution and is transported from the vicinity of the boron, or the boron is leached from the container. In the former case, the fissile material must then be precipitated in a shape and mass to promote critical conditions. Thus sufficient water must be available to corrode the container and dissolve either the boron or the fissile material so that it can be transported. Water may also be important for promoting critical conditions because water in and around fissile material can slow (moderate) neutrons from other fissions (splitting of atoms) such that the neutrons are more readily absorbed and cause a fission in the fissile material. (Other moderators are theoretically possible, though the assembly of a natural reactor with other moderators is unlikely.)

Fault Tree Analysis. To obtain a rough estimate of the probability of assembly of fissile mass, the design features of the containers, the orientation of waste emplacement, and the repository layout were combined in a fault tree. Because water is essential for producing conditions conducive to a criticality by significantly changing the container, allowing assembly of the fissile material, and possibly moderating the nuclear reaction, the probability assigned to climatic change and subsequent percolation is crucial in the fault tree. The fault tree flatly assumed that enough water to create a criticality would be introduced as a result of a climatic change and subsequent percolation through connecting cracks or “weeps” in the tuff. Based on this crucial assumption, a water-moderated criticality involving at least one waste container over a duration of 10,000 yr had a probability above the regulatory cutoff (10⁻⁴) for which a scenario can be ignored.

Detailed Process Analysis. The advantage of a fault tree is that this structured approach can quickly identify events and situations that may lead to a criticality in a repository. It was this advantage that prompted its use, as described above, in order to develop hypotheses of behavior. A disadvantage is that the primary components of a tree for a geologic repository are usually natural processes. The probability of occurrence of these processes may be high, but whether they induce adverse conditions and promote a critical condition is not known; thus, evaluating the overall probability of a critical event is difficult. Hence, the 1994 PA provided a more detailed evaluation of the criticality that included required processes and the interplay of the controlling parameters (such as fissile mass and power output) within the (a) geologic barrier (and corresponding rock characteristics and water percolation), (b) repository (and corresponding layout and waste emplacement orientation), and (c) waste form (and corresponding alteration) (Figure ES-13).

In an unpressurized system, the multiplication factor (kₑff) was found to be extremely sensitive to minute changes in water saturation (Figure ES-13) that, in turn, was very sensitive to changes in temperature. This situation provided strong negative feedback that rapidly shut down a nuclear chain reaction once started. For the purposes of better understanding the occurrence of a criticality, the range of water available for corrosion of protective layers and alteration and dissolution of waste was manually coupled with the results of the performance assessment. Results showed that a nuclear reaction could not be maintained at a power level associated with aqueous accidents (about 10¹⁵ fissions per incident) in a natural unpressurized system. 

Table ES-13 summarizes the information gathered about critical conditions in or near containers.

†††Although one might assume that the boron is more soluble than uranium, borated stainless steel may incorporate boron as a metal boride, (e.g., chromium boride), which will release boron extremely slowly. Thus uranium may be removed from the container before boron.

††††One can reasonably conjecture that the changes in saturation could also limit the highly exothermal oxidation of uranium metal by water vapor in the ATR and N-Reactor fuels. Although oxygen may also promote oxidation of uranium, it could be limited as a reactant, as well, as explained in Section ES.2.4, Repository Design.
Figure ES-13. Hypothesized situation leading to a critical mass near a container and the hypothesized power output, based on aqueous accidents, resulting in a calculated time history of temperature, saturation, and multiplication factor \((k_{\text{eff}})\). The periodicity depends on overshoot of temperature, which may not occur if the approach to criticality is asymptotic over a very long time. However, minor natural fluctuations of water saturation in a real system could disrupt this asymptotic approach to a critical condition. The extreme sensitivity of \(k_{\text{eff}}\) to minute changes in saturation rapidly shuts down a nuclear reaction, if ever started.
Table ES-13. Summary of Information Gathered about Critical Conditions in or Near Containers

Consequences of Critical Conditions assuming High Moderation and Low Reactivity

1. A steady-state reactor limited to 1 kW (-1 mg $^{235}$U) to prevent boiling of water in a container would generate about $10^{25}$ fissions over 10,000 yr.

2. The percentage increase in the total inventory and the inventory of most of the radionuclides from fission, should one container go critical and produce $10^{25}$ fissions (similar to documented aqueous accidents) over 10,000 yr, is insignificant. Even $10^{28}$ fissions in about 289 containers with highly enriched uranium would be only 1% of the $-10^{30}$ fissions represented by a 12,060-MTHM repository containing spent fuel with burnup of 40,000 MWh/MTU.

3. Using a 2-phase flow composite-porosity model and between 1 and 10 kg of fissile material, the forced startup of a natural reactor would potentially operate for between 1 and 1000 days (for $10^{21}$ and $10^{19}$ fissions, respectively) before the water saturation dropped by 15%. The system took between 70 and 200 days to recover to the initial saturation.

4. In the slow fission process envisioned to occur in a repository, gaseous fission products such as $^{85}$Kr (with the longest half-life of 10.78 yr) decay more rapidly than they are transported to the regulated boundary in the first 500 yr, but could escape after 1000 yr (see Figure ES-9). However, diffusion into the atmosphere would greatly diminish the doses.

5. The effect of additional heat generation from a criticality condition on regional liquid flow is also insignificant.

Probability of Critical Conditions

6. A preliminary fault tree analysis showed that the possibility of a critical event could not be readily dismissed (ignored as a scenario).

7. Assuming a composite-porosity model with climate change, concentration of the infiltration, and the disposal containers of 20-mm Incoloy Alloy 825 and 100-mm carbon steel, a median value of 30 and a mean of 700 kg of fissile material would be transported from all 475 MPCs containing spent nuclear fuel within 10,000 yr.

8. The operation of a natural unpressurized waste reactor is extremely sensitive to water saturation. Because $k_{eff}$ is changed by any slight change in saturation, the nuclear chain reaction is immediately shut down.

9. Using the assumptions of these preliminary nuclear dynamics calculations, a large pressure rise is not feasible in a partially saturated, unpressurized natural reactor because of the sensitivity of $k_{eff}$ to the saturation.

10. Between $10^1$ to $10^7$ times more ATR uranium metal than borides (e.g., chromium-boride) is altered inside containers.

Of recent concern among the geological repository community has been the possibility of a dry, silica-moderated system that could experience positive feedback (Broad, 1995). The reported study primarily evaluated the energy released from a preassembled, highly critical mass of fissile material. The work performed has been described as rough (Johnson, 1995) and is subject to criticism (e.g., the tuff is far from being pure silica and has many neutron absorbers). In addition, given that the most important feature of a criticality analysis is a determination of initial conditions, as mentioned above, the silica-moderated hypothesis lacks a credible process within a geologic repository to assemble the fissile mass in a highly supercritical state without first passing through a more credible but less energetic critical state. For example, although silicon-dioxide has the classical characteristics of a moderator, it is very inefficient, requiring large masses of fissile material to achieve a critical assembly. The isolation of such large masses of fissile material requires significant volumes of groundwater to either transport the absorbers away from the fissile material or to transport fissile material to an assembly location with an optimal mixture of the fissile material and the silica (diffusion of the fissile material into the tuff would not occur within a reasonable time scale). Thus, given these initial conditions, a startup of the nuclear reaction with water as the moderator would be favored, as studied in the 1994 PA.

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ES.3 Guidance to the Project

The findings listed in the previous section provide the basis for the following guidance to DOE on the potential compliance determination for the waste disposal option under study, i.e., direct disposal of spent nuclear fuel and high-level waste, with minimal treatment or conditioning, in a two-layer disposal container emplaced in a repository in unsaturated tuff.

ES.3.1 General Guidance for Waste Treatment Program

In answer to the question regarding what decisions about treatment can be made by DOE before a repository type and site have been identified, the findings suggest that most decisions on treatment or conditioning should wait until a repository type and site are known. This conclusion is based on the finding that minimal treatment of the waste types was sufficient to remain within the reference range defined by the DOE borosilicate glass and the pressurized water reactor fuel, even though a robust (300 to 1000 yr) container was not used in the tuff disposal system. The results also showed compliance with current regulations but the absolute position of results is uncertain because full characteristics of the repository and waste are incomplete and the form of future regulations is unknown. This same conclusion was reached for the salt and granite disposal systems in the 1993 PA. This recommendation takes into consideration not only the results of the 1993 and 1994 PAS, but also the expectation of possible changes to the regulations. For example, policies could move from imposing general regulations toward developing site-specific criteria, as suggested in the Energy Policy Act of 1992, Section 801.

Also, it should be noted that the criticality calculations are preliminary and so provide merely an indication of the low likelihood of a critical occurrence. Nevertheless, the calculations appear to support the belief that it would be difficult to create conditions that would provide enough water to (1) corrode the containers (both those with and without a corrosion-resistant material), (2) remove neutron absorbers or uranium, and (3) moderate a nuclear chain reaction within the repository. In addition, a criticality may not be technically disruptive to repository performance.

Thus, it is recommended that only those decisions that relate to compliance with current court orders or regulations for storage of high-level waste and spent fuel should be made before a repository site is known.

These findings do not mean that characterization of waste is sufficient or complete. Specific recommendations on data needs such as characterization of the waste types, experiments on solubility of radionuclides, and corrosion studies of potential containers that should be pursued to support future disposal are listed in Section ES.3.3. Even if credit on the license application is taken only for a robust container, some preliminary characterization is expected to be required to assure the regulator that DOE has a rudimentary understanding of its waste. Characterization of the waste would also provide a basis for future treatment of waste, if treatment is considered necessary for acceptance of the waste into a repository.

Little work has been done to establish repository post-closure criticality safety requirements and policy. Thus, there is not as yet a consistent or coherent approach to what technical (risk-based) assumptions can or should be applied in a criticality. Furthermore, methodologies necessary to deal with technical analysis and evaluations (PAs) assignment of risk quantification, probabilistic risk assessments, and criticality calculations need some degree of standardization. Because commercial-type fuel and highly enriched spent fuel have many characteristics in common, cooperative efforts between Office of Environmental Management (DOE-EM), which is responsible for the DOE spent fuel, and the Office of Civilian Radioactive Waste Management (DOE-RW), which is responsible for commercial spent fuel, could be mutually beneficial.

*****The use of a robust container may not be important for the individual dose requirement; rather, a container that fails over a wide range of times may be more useful.
ES.3.2 Guidance for Waste Acceptance Criteria

Assuming a total system evaluation, which is consistent with the criteria of 40 CFR 191, the following guidance is given for any waste acceptance criteria.

First, for a repository in unsaturated tuff, disposal of currently existing DOE spent fuel and high-level waste with minimal treatment or conditioning in containers proposed by the Yucca Mountain Project in 1994 will likely show compliance with the Containment Requirements of 40 CFR 191.13. The results also show that waste disposal in a container that does not meet the requirements of 10 CFR 60.113 could still satisfy compliance requirements of 40 CFR 191.13 for 10,000 yr, individual dose requirements in 40 CFR 191.15 for 10,000 yr, and international guidelines on dose using a deterministic calculation up to 100,000 yr at a location 5 km from the waste.

Second, no characteristic of the spent fuel owned by the DOE would contribute to a repository in tuff, salt, or granite being less likely to comply with existing regulations than if it contained other spent fuel types, based on the results of the 1993 and 1994 PAs. Issues under study for the DOE-owned waste were criticality and the effects of conditioning. The issue of criticality is important for the DOE-owned spent fuel; however, criticality is potentially important for commercial spent fuel as well (current regulations provide no guidance on the issue of criticality). However, the effects of critical conditions in a repository are not perceptible although the calculations regarding criticality are preliminary and thus cannot be relied upon for final guidance. Also, although conditioning does not appear to be necessary for DOE-owned spent fuel with regard to compliance with regulations regarding geologic disposal, some conditioning of the N-Reactor and Advanced Test Reactor fuel might be necessary to improve safety in handling and transportation to the repository; however, this possibility was not studied here.

Third, the combined results of the 1993 and 1994 PAs indicate that the acceptability of the waste form or, alternatively, the need for a container that meets the requirements of 10 CFR 60, in order to meet the overall requirements of 40 CFR 191, is sensitive to the geologic medium of the disposal system (i.e., the type of host rock). Thus, preliminary waste acceptance criteria must clearly indicate the relationship between acceptance and the type of geologic medium in which the repository is located.

ES.3.3 Recommendations for Future Analysis and Data Acquisition

The use of performance assessments as a tool during the development and demonstration of the capability to dispose of spent nuclear fuel and high-level waste offers distinct advantages. First, a performance assessment is a valuable and low-cost tool, relative to actual implementation of waste treatment and disposal operations, that responds to the spirit of the Nuclear Waste Policy Act (with amendments in 1987) (NWPA, 1983; NWPAA, 1987). Second, a performance assessment is the primary means by which the DOE will demonstrate compliance with long-term environmental regulations and, therefore, should be an essential component of any development program. The INEL Spent Fuel and Waste Management Technology Development Program is unique in recognizing the benefits of a performance assessment early in the program.

In addition to providing guidance to the project, this performance assessment serves to inform both the analysis team and decisionmakers who commissioned the study about (a) the type of model development and analysis that should be conducted in the future and (b) the scientific information that must be gathered, either from the literature or investigative studies, for use in models selected for future performance assessments. Recommendations about model development and analysis can be acted upon by the analysis team; recommendations about gathering additional data may require input from throughout the DOE complex. Both types of recommendations are presented in Tables ES-14 and ES-15.

Recommendations for Future Model Development and Analysis. The recommendations for future model development and analysis are as follows (Table ES-14).
Table ES-14. Recommendations for Future Model Development and Analysis

Source-Term and Repository Modeling Development and Analysis

1. Develop the capability to run simulations that transport oxygen within the repository in order to use both oxic and anoxic corrosion rates and solubility values in the source term model.

2. Continue computations that estimate solubilities of radionuclides in solutions saturated with and without silica.

3. Include an increase of surface area as encapsulating fuel or glass matrix alters over time.

4. Evaluate the effects of the steel and concrete used during the operational phase and left in the repository after closure.

5. Evaluate whether reactants such as oxygen and water are limited significantly to reduce the rate of oxidation of the uranium metal in some types of DOE spent fuel.

6. Develop a version of the source term submodel that can be run separately from the fluid flow and transport codes.

Suggested Criticality Modeling Development and Analysis

7. Examine the corrosion and leaching of borated stainless steel or other material versus the removal of uranium from a container.

8. Examine more uncoupled events to develop an understanding of the feasibility of the critical event in a moderated but unpressurized assembly of fissile material.

9. Continue to examine more of the conditions and assumptions hypothesized in the fault tree for criticality event in or near a waste container.

10. Examine assemblies with larger amounts of fissile material than were used in the 1994 PA to see whether sensitivity to minute changes in saturation is reduced.

11. Examine potential options to lower the probability of critical conditions occurring within the repository.

12. Examine the possibility and consequences of a critical condition at the surface after an exploratory borehole has penetrated the repository and brought fissile material to the surface.

13. Develop the capability to include precipitation of fissile material in transport calculations.

Suggested Geologic Barrier Analysis

14. Compare results assuming a full dual-permeability model to the composite-porosity simplification.

15. Evaluate the possibility of an intrusion borehole enhancing flow above the repository, to a container, and then to the water table.

16. Examine the influence of a weeps conceptual model for flow within tuff.

17. Explore possible ranges for a rise in the water table and its influence on the behavior of DOE spent fuel and high-level wastes.

Suggested Overall Analysis

18. Re-examining the salt and granite disposal options to provide more consistency between the 1993 and 1994 PAs, specifically using the improved source term model from the 1994 PA, and examining the potential for critical conditions developing within or near containers.

19. Compare the advantages and disadvantages of salt, granite, and tuff disposal systems.

20. Evaluate individual doses from the disposal system beyond 10,000 yr.

21. Assess the performance of an unsaturated tuff disposal system containing DOE high-level waste and spent fuel from both public utilities and DOE, since temperature has a large influence on corrosion rates.

22. Evaluate the feasibility of assigning values and ranks to different investigative programs to help allocate program resources.
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Table ES-15. Recommendations for Data Acquisition

1. Pursue better characterization of the current spent fuel and high-level waste types, specifically, data on spent fuel with regard to failed protective coatings.
   - Estimates of the likelihood of failure of protective coatings on spent fuel in the environment of the repository during the regulatory period.
   - The chemical reactivity of the N-Reactor and Advanced Test Reactor fuels and plans for stabilizing and treating the fuel for storage.
   - The amount of $^{99}$Tc in the calcine high-level waste should be verified.

2. Seek to confirm or develop consensus of alteration rates used in the performance assessment, perform a more thorough literature search and support experiments to obtain temperature-dependent data.
   - For the gas consumption/generation submodel, generalized corrosion rates specific to conditions expected in the tuff repository are needed for the container materials.
   - For the source term submodel, more reliable information about susceptibility and data on failure modes (specific to conditions expected in the tuff repository are needed for the container materials.

3. Continue to obtain better sorption coefficients used for materials inside and outside the waste parcel (including corrosion products), materials in the unsaturated zone, and materials in the saturated zone.

4. Continue to seek more in-depth information on solubilities of important elements.
   - For the tuff disposal system, these include technetium, iodine, uranium, and neptunium.

5. For the evaluation of critical conditions, obtain defensible release rate of boron from borated steel.

Suggested Source Term and Repository Model Development and Analysis. A possible disadvantage of an unsaturated repository, such as tuff, is that the chemical environment is oxic and thus corrosion rates of protective materials and solubilities of neptunium (Np) and some other actinide radionuclides are greater than in a saturated repository.

In the 1994 PA, we assumed that the area around the repository always remained oxic even though large quantities of iron were present, which would be expected to be reducing, especially within the container. Thus, future work should develop the capability to run simulations that transport oxygen within the repository in order to use both oxic and anoxic corrosion rates and solubility values in the source term model. This capability would permit a more thorough evaluation of whether reactants such as oxygen and water are limited in reactions within or near the container and thus limit the oxidizing rate of uranium metal, the corrosion and alteration of materials, and the solubility of radionuclides. This capability would permit an evaluation of whether materials such as iron, which promote a reducing chemical environment, are useful in terms of waste disposal. Doing so would also allow an examination of the use of filler in the container to further promote a reducing environment.

Several factors affect the solubility of radionuclides, such as (a) the availability of oxygen, (b) the concentration of dissolved silica, (c) pH, (d) precipitation kinetics, and (e) temperature. Because these factors are expected to be markedly different in the far field and within each type of waste parcel, the 1994 PA included two sets of solubilities for uranium, neptunium, and plutonium. Higher solubilities were used for the inside of the container, because silica in the percolating waters there may be removed by reaction with corrosion products, the temperature is higher than outside the container, and there may be more time for formation of more ordered precipitates. Lower solubilities were used outside the container because percolating water there is saturated with silica, the temperature is lower than inside the container, and there is more time for recrystallization of precipitates. This condition, which represents a potential advantage for a tuff repository, needs to be examined in more detail and for other radionuclides. In addition, future analysis should examine other effects such as the availability of oxygen (Eh) inside and outside the container with the effect of silica saturation, pH, precipitation kinetics, and temperature on radionuclide solubility.

The surface area exposed to percolating water can greatly increase as the waste matrix alters, and so a minor improvement in the source term model would be to include an increase of surface area as encapsulating fuel or glass matrix alters over time.

Besides the waste and containers, a large mass of materials such as concrete and steel could be necessary in the operation of the repository. This material has the potential to alter the chemical environment, and thus future studies
should evaluate the effects of the steel and concrete used during the operational phase and left in the repository after closure.

The chemical reactivity of the uranium metal in the repository is a potential concern. It may be that reactants are limited sufficiently in a deep geologic repository to maintain a controlled oxidation, much as the nuclear chain reaction was found to be controlled in this year's performance assessment; however, future studies should evaluate whether reactants such as oxygen and water are limited significantly to reduce the rate of oxidation of the uranium metal in some types of DOE spent fuel.

Many of the results reported herein are directly related to the output of the gas consumption/generation and source term models, which presently run as submodels within the fluid flow and transport codes. To respond rapidly to design and "what if" questions about the waste forms and to perform sensitivity/uncertainty analyses, it would be useful to develop a version of the source term submodel that can be run separately from the fluid flow and transport codes.

Suggested Criticality Modeling Development and Analysis. Although much was accomplished in the 1994 PA with regard to criticality, future studies need to perform more calculations to adequately model processes that would create conditions conducive to a criticality within the repository and its consequences. A key analysis area that needs additional study is to examine the corrosion and leaching of borated stainless steel (or other borated material) versus the removal of uranium. Although some calculations were done, a detailed analysis of the corrosion of borated stainless steel in the canister inner baskets was not possible given the scope of the 1994 PA. This modeling is necessary to identify whether borated material is leached away from a canister over a 10,000-yr period in amounts substantial enough to be of concern. (Initial calculations indicated that uranium migrated before the boron; see Chapter 10.) Included in this work would be an analysis to determine the solubility of boron in the tuff groundwater and corresponding material transport. Initially, this modeling should examine more uncoupled events to develop an understanding of the feasibility of the critical event in a moderated but unpressurized assembly of fissile material before coupling currently existing nuclear reactor, lumped-parameter codes with current two-phase flow codes to evaluate dynamics of critical conditions. This analysis can first continue to examine more of the conditions and assumptions hypothesized in the fault tree for criticality event in or near a waste container. However, a coupling of fluid flow and nuclear physics can be done without much more effort and should also be attempted because the end result may be more informative.

While the probability of assembling large amounts of fissile material is low, future studies should perform coupling calculations to examine assemblies with larger amounts of fissile material than were used in the 1994 PA to see whether sensitivity to minute changes in saturation is reduced.

Because the assembly of sufficient fissile material and water for moderation is difficult in the geologic setting, the addition of a few features in the container or repository could further reduce the probability of a critical condition. Hence, future studies should examine potential options to lower the probability of critical conditions occurring within the repository such as adding filler (internal backfill) within the container to reduce the volume of water available to moderate nuclear reactions.

An additional area is to examine the possibility and consequences of a critical condition at the surface after an exploratory borehole has penetrated the repository and brought fissile material to the surface. Because bringing fissile material to the surface already represents a release, this situation was not examined in the 1994 PA. However, the critical event should probably be quantified to some extent such that the additional hazards from a criticality can be compared to the hazard of releasing the contents of a spent fuel container at the surface.

A final area to be explored is to develop the capability to model precipitation of fissile material in transport calculations. Incorporating this capability into current models would be relatively easy; the more difficult aspect would be constructing numerical models with fine grids, yet reasonable run times, and the proper correlation of solubilities within and outside the container.

Suggested Geologic Barrier Analysis. Movement of water in and around the repository is an important phenomenon to continue to explore in future studies. The 1994 PA used a composite-porosity simplification in the dual-per-
measurability conceptual model. A useful step would be to compare results assuming a full dual-permeability model to the composite porosity simplification. In addition, the complex behavior in non-isothermal two-phase flow and transport in a heterogeneous fractured porous medium may require models yet to be developed. Involvement in this model development should occur. Enhanced flow from an intrusion borehole was assumed negligible in the 1994 PA because the fracture network was considered to be already permeable. However, scoping calculations should be performed to evaluate the possibility of an intrusion borehole enhancing flow above the repository, to a container, and then to the water table. Furthermore, future studies should examine the influence of alternative conceptual models of fluid flow, specifically, examine the influence of a weeps conceptual model for flow within tuff. The 1994 PA assumed the influence of a rise in the water table in the first 10,000 yr would be negligible because the change in infiltration was minor over this same regulatory period. However, performing calculations out to 100,000 yr will increase the influence of climatic change and thus the possibility of a rise in the water table. Certainly a large regional model would be necessary to quantify the potential rise in the water table, which might classify this study within the scope of work performed by a specific, potential repository. However, future studies should at least explore possible ranges for a rise in the water table and its influence on the behavior of DOE spent fuel and high-level wastes.

**Suggested Overall Analysis.** Several improvements or changes in the overall analysis should be pursued in future studies. These include re-examining the salt and granite disposal options to provide more consistency between the 1993 and 1994 PAs, specifically using the improved source term model from the 1994 PA, and examining the potential for critical conditions developing within or near containers in these disposal systems. A natural addition would be to compare the advantages and disadvantages of salt, granite, and tuff disposal systems. Such a comparison can be only qualitative now because of the many changes that occurred between the 1993 and 1994 PAs.

The international community continues to carry out its simulations well beyond 10,000 yr. Although doing so is not specifically required in the United States, the potential benefits of comparing our results with those from other nuclear waste disposal programs throughout the world suggest the need to evaluate individual doses from the disposal system beyond 10,000 yr.

Because of temperature's strong influence in a tuff disposal system on the corrosion rates of the disposal container and other protective layers of the spent fuel and high-level waste, specific design goals for temperature are needed for any potential repository. Before these goals become available, it may be informative to assess the performance of an unsaturated tuff disposal system containing DOE high-level waste and spent fuel from both public utilities and DOE.

An analysis area that needs improvement is for the results of a performance assessment to become more directly applicable to decisions about programmatic directions. Thus, future studies should evaluate the feasibility of assigning values and ranks to different investigative programs to help allocate program resources. For example, the value of characterizing the fraction of particles with failed silicon carbide coatings or claddings on spent nuclear fuel might be ranked against the value of experiments on the solubility of radionuclides in water saturated with silica.

**Recommendations for Data Acquisition.** Because a need exists for the best, credible data possible in many areas, especially with regard to future licensing applications, specific recommendations for data acquisition with regard to evaluating the performance of DOE spent fuel and high-level waste in a geologic repository are as follows (see also Table ES-15).

The waste treatment program should pursue better characterization of the current spent fuel and high-level waste types, specifically, data on spent fuel with regard to failed protective coatings (e.g., fraction of particles with failed silicon carbide coatings is an important model parameter for the graphite spent fuel); estimates of the likelihood of failure of protective coatings on spent fuel in the environment of the repository during the regulatory period (e.g., likelihood of failure of silicon carbide coatings of fuel particles in the graphite fuel), and the chemical reactivity of the N-Reactor and Advanced Test Reactor fuels and plans for stabilizing and treating the fuel for storage. Based on current regulations, the amount of $^{14}$C present and its location (e.g., cladding or UO$_2$ matrix) in the spent fuel owned by DOE would be pertinent; however, anticipated changes to a standard based on individual doses would eliminate the need for this specific information. The amount of $^{99m}$Tc in the calcine high-level waste should be verified before use in future analysis.
The waste treatment program should also seek to confirm or develop consensus of alteration rates used in the performance assessment, specifically, perform a more thorough literature search and support experiments to obtain temperature-dependent data. For the gas consumption/generation submodel, generalized corrosion rates (of, for example, 304L stainless steel, carbon steel, Inconel Alloy 825, and zircaloy) specific to conditions expected in the tuff repository are needed for the container materials. For the source term submodel, more reliable information about susceptibility and data on failure modes (e.g., stress-corrosion cracking, pitting, crevice corrosion, delayed hydrogen-induced failure, and aging reactions) specific to conditions expected in the tuff repository are needed for the container materials to adequately assess the longevity of the barrier capabilities of the container.

For credit to be taken for the potential sorption of radionuclides, a waste treatment program should continue to obtain (both through literature and experiments) better sorption coefficients used for materials inside and outside the waste parcel (including corrosion products), materials in the unsaturated zone, and materials in the saturated zone.

Even accounting for numerous protective layers in spent nuclear fuel, elemental solubility remains important data. Although direct measurements may not be feasible because (1) equilibrium conditions are difficult to reach in lab experiments and (2) the experiments are costly and time consuming because of environmental, safety, and health issues, the waste treatment program needs to continue to seek more in-depth information on solubilities of important elements. For the tuff disposal system, these include technetium, iodine, uranium, and neptunium.

For the evaluation of critical conditions, it is important to obtain defensible release rate of boron from borated steel.

Some of the recommendations found in Tables ES-14 and ES-15 were also raised in the 1993 PA. However, many of the recommendations of the 1993 PA were acted upon this year, including:

1. Conducting a performance assessment of a repository in unsaturated tuff, for disposal of spent nuclear fuel and high-level waste, continuing the practice of using different levels of simplification to evaluate consequences. (This assessment required developing a much more versatile source term model and incorporating a composite-porosity model in existing software. The latter improvement allowed two-phase flow and transport modeling in a composite-porosity continuum to be modeled directly in the 1994 PA.)

2. Examining performance criteria other than 40 CFR 191. (In the 1994 PA, we examined the waste package and engineered barrier requirements in 10 CFR 60.135, the individual dose requirements of 40 CFR 191 and international dose guidelines [Volume 3, Appendix D] in addition to the containment requirements of 40 CFR 191.13.)

3. Performing calculations to determine whether the occurrence of critical conditions in a repository would be disruptive to repository performance. (In the 1994 PA, the calculations have covered many different criticality conditions.)

4. Modeling diffusion of contaminants through rubble contacting containers.

5. Acquisition of some data on alteration rates of glass-ceramic matrix and percentage of radionuclides trapped in glass and ceramic phases.

6. Performing geochemical calculations and literature review to obtain a better understanding of element solubility within the repository. (A side benefit was that for criticality calculations, we determined likely mineral phases of uranium that would be precipitated in the repository environment.)

7. Selecting the MPC waste container design of the Yucca Mountain Project as a “reference design” for the performance calculations, although a conceptual LWT container for vertical emplacement was also used.
ES.4 References


