Appendix I

Environmental Consequences of Long-Term Repository Performance

TABLE OF CONTENTS

Section Page

I. Environmental Consequences of Long-Term Repository Performance ... I-1

I.1 Introduction ... I-1

I.2 Total System Performance Assessment Methods and Models ... I-2
I.2.1 Features, Events, and Processes ... I-4
I.2.1.1 Tectonic Activity ... I-6
I.2.1.2 Erosion/Denudation ... I-6
I.2.1.3 Periglacial Effects ... I-7
I.2.1.4 Glacial and Ice Sheet Effects ... I-8
I.2.1.5 Hydrostatic Pressure on Container ... I-8
I.2.1.6 Soil and Sediment Transport ... I-8
I.2.1.7 Seismic Damage to Waste Packages ... I-9
I.2.2 Unsaturated Zone Flow ... I-9
I.2.3 Engineered Barrier System Environments ... I-11
I.2.4 Waste Package and Drip Shield Degradation ... I-14
I.2.5 Waste Form Degradation ... I-15
I.2.6 Engineered Barrier Transport ... I-16
I.2.7 Unsaturated Zone Transport ... I-16
I.2.8 Saturated Zone Flow and Transport ... I-17
I.2.8.1 Saturated Zone Flow ... I-18
I.2.8.2 Saturated Zone Transport ... I-18
I.2.9 Biosphere ... I-19
I.2.10 Volcanism ... I-20
I.2.11 Human Intrusion ... I-21
I.2.12 Nuclear Criticality ... I-21
I.2.13 Atmospheric Radiological Consequences ... I-21

I.3 Inventory ... I-21
I.3.1 Inventory for Waterborne Radioactive Materials ... I-21
I.3.1.1 Radionuclide Inventory Used in the Model of Long-Term Performance for Proposed Action ... I-25
I.3.1.2 Radionuclide Inventory Used in the Model of Long-Term Performance for Inventory Module 1 ... I-25
I.3.1.3 Radionuclide Inventory Used in the Model of Long-Term Performance for Inventory Module 2 ... I-26
I.3.2 Inventory for Waterborne Chemically Toxic Materials ... I-28
I.3.3 Inventory for Atmospheric Radioactive Materials ... I-29

I.4 Extension of TSPA Methods and Models for EIS Analysis of Long-Term Performance ... I-29
I.4.1 Methodology ... I-30
I.4.2 Assumptions ... I-31
I.4.3 Use of Computer Software and Models ... I-32
I.4.4 Modifications to the TSPA–Site Recommendation and Supplemental Science and Performance Analysis Models ... I-32
I.4.4.1 Modifications to FEHM Particle Tracker Input and Output ... I-33

Section Page
I.4.4.2 Estimation of the Thermal Profiles and Infiltration for the Lower Block I-34
I.4.4.3 Saturated Zone Breakthrough Curves I-35
I.4.4.4 Modification to the Waste Package Degradation Model I-37
I.4.4.5 Early Waste Package Failure I-38
I.4.4.6 Biosphere Dose Conversion Factors for the 40 CFR 197 Reasonably Maximally Exposed Individual I-39
I.4.4.7 Igneous Activity Scenario I-39
I.4.4.8 Human Intrusion Scenario I-41
I.4.5 Extension of Groundwater Impacts to Other Distances I-42
I.4.5.1 Locations for Assessing Postclosure Impacts to Human Health I-42
I.4.5.2 Scaling Approach I-45
I.4.5.3 Scaling Factors for Dose or Water Quality Concentrations at Longer Distances I-46

I.5 Waterborne Radioactive Material Impacts I-46
I.5.1 Waste Package Failure I-47
I.5.2 Annual Individual Dose for 10,000 Years After Closure I-48
I.5.3 Annual Individual Dose for 1,000,000 Years After Closure I-48
I.5.3.1 Annual Individual Dose for the Proposed Action Inventory, Higher- and Lower-Temperature Repository Operating Modes I-49
I.5.3.2 Annual Individual Dose for Inventory Module 1, Higher-Temperature Repository Operating Mode I-50
I.5.3.3 Annual Individual Dose for Inventory Module 2, Higher-Temperature Repository Operating Mode I-50
I.5.3.4 Annual Individual Dose for Igneous Activity Scenario, Higher- and Lower-Temperature Repository Operating Modes I-50
I.5.3.5 Annual Individual Dose for the Human Intrusion Scenario I-51
I.5.4 Comparison to Groundwater Protection Standards I-51

I.6 Waterborne Chemically Toxic Material Impacts I-52
I.6.1 Screening Analysis I-52
I.6.1.1 Maximum Source Concentrations of Chemically Toxic Materials in the Repository I-52
I.6.1.2 Further Screening for Chemically Toxic Materials I-54
I.6.2 Bounding Consequence Analysis for Chemically Toxic Materials I-54
I.6.2.1 Assumptions I-55
I.6.2.2 Surface Area Exposed to General Corrosion I-56
I.6.2.3 General Corrosion Rates I-59
I.6.2.4 Release Rates I-60
I.6.2.5 Summary of Bounding Impacts I-60

I.7 Atmospheric Radioactive Material Impacts I-62
I.7.1 Carbon-14 Releases to the Atmosphere I-63
I.7.2 Atmosphere Consequences to the Local Population I-64
I.7.3 Screening Argument for Radon I-66

References I-85

LIST OF TABLES

Table — Page

I-1 Average net infiltration rates over the unsaturated zone flow and transport model domain for the present-day, monsoon, and glacial transition climate states ... I-10
I-2 Average net infiltration rates over the unsaturated zone flow and transport model domain for full-glacial climate states ... I-10
I-3 Poisson probabilities for improper heat treatment of waste packages ... I-15
I-4 Modeled number of idealized waste packages by category type for the abstracted inventories of the Proposed Action, Inventory Module 1, and Inventory Module 2 ... I-25
I-5 Abstracted inventory of radionuclides passing the screening analysis in each idealized waste package for commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste for Proposed Action, Inventory Module 1, and Inventory Module 2 ... I-26
I-6 Greater-Than-Class-C low-level waste volumes by source ... I-27
I-7 Abstracted inventory of radionuclides passing the screening analysis in each idealized waste package for Greater-Than-Class-C and Special-Performance-Assessment-Required waste under Inventory Module 2 ... I-28
I-8 Inventory of chemical materials placed in the Proposed Action repository ... I-29
I-9 Carbon-14 gaseous inventory from commercial spent nuclear fuel ... I-30
I-10 Average values of the dietary and lifestyle attributes for the RMEI ... I-39
I-11 Biosphere dose conversion factors for the RMEI for the groundwater release and the volcanic release exposure scenarios ... I-40
I-12 Groundwater impact distance scale factors for 99-percent captured mass, longitudinal dispersivity 100 meters, horizontal dispersivity 10 meters, and vertical dispersivity 0.1 meters ... I-47
I-13 Peak mean annual individual doses for analyzed inventories, scenarios, and temperature operating modes ... I-48
I-14 Peak 95th-percentile annual individual doses for analyzed inventories, scenarios, and temperature operating modes ... I-49
I-15 Peak mean gross alpha activity for analyzed inventories, scenarios, and temperature operating modes ... I-53
I-16 Peak mean total radium concentration for analyzed inventories, scenarios, and temperature operating modes ... I-53
I-17 Peak mean annual whole body or organ dose for the sum of all beta- and photon-emitting radionuclides during 10,000 years after closure for analyzed inventories, scenarios, and temperature operating modes ... I-53
I-18 Total exposed surface area of the Alloy-22 outer layer of all waste packages under the Proposed Action inventory ... I-56
I-19 Total exposed surface area of the Alloy-22 rails for all drip shields under the Proposed Action inventory ... I-56
I-20 Total exposed surface area of the Alloy-22 outer layer of all waste packages for the Module 1 inventory ... I-57
I-21 Total exposed surface area of the Alloy-22 rails for all drip shields for the Module 1 inventory ... I-57
I-22 Total exposed surface area of the Alloy-22 outer layer of all waste packages for the Module 2 inventory ... I-57
I-23 Total exposed surface area of the Alloy-22 rails for all drip shields for the Module 2 inventory ... I-58

Table		Page
I-24	Total exposed surface area of the Alloy-22 components for all emplacement pallets under the Proposed Action, Module 1, and Module 2 inventories	I-58
I-25	Total exposed surface area of the stainless-steel 316NG components for all emplacement pallets under the Proposed Action inventory	I-59
I-26	Total exposed surface area of the stainless-steel 316NG components for all emplacement pallets for the Module 1 inventory	I-59
I-27	Total exposed surface area of the stainless-steel 316NG components for all emplacement pallets for the Module 2 inventory	I-59
I-28	Bounding mass release rates from Alloy-22 and stainless-steel 316NG components from general corrosion for the Proposed Action	I-60
I-29	Bounding mass release rates from Alloy-22 and stainless-steel 316NG components from general corrosion for Module 1	I-60
I-30	Bounding mass release rates from Alloy-22 and stainless-steel 316NG components from general corrosion for Module 2	I-61
I-31	Bounding concentrations of waterborne chemical materials of concern compared to Maximum Contaminant Levels Goals	I-61
I-32	Summary of intake of waterborne chemical materials of concern based on maximum bounding concentrations listed in Table I-31 compared to Oral Reference Doses	I-62
I-33	Meteorologic joint frequency data used for Yucca Mountain atmospheric releases	I-65

LIST OF FIGURES

Figure		Page
I-1	TSPA model	I-68
I-2	Development of abstracted inventory for TSPA-FEIS	I-69
I-3	Approximate configuration of the potential Yucca Mountain Repository	I-70
I-4	The four saturated zone capture regions in relation to the primary repository block for the Proposed Action	I-71
I-5	The four saturated zone capture regions in relation to the primary and lower repository blocks for Inventory Modules 1 and 2	I-72
I-6	Outline of the Lower Block showing the locations of the 51 particle-tracking nodes	I-73
I-7	Southernmost boundary of the controlled area and the accessible environment	I-74
I-8	Waste package failure curves for 300 probabilistic simulations for the Proposed Action inventory; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations	I-75
I-9	Cladding failure profile for the Proposed Action inventory	I-75
I-10	Comparison plot of the total mean annual individual dose at the RMEI location under the higher-temperature and lower-temperature operating modes for the Proposed Action inventory, nominal scenario	I-76
I-11	Total and individual radionuclide mean annual dose to an individual at the RMEI location for the higher-temperature operating mode for the Proposed Action inventory, nominal scenario	I-76
I-12	Total annual individual dose at the RMEI location for 300 probabilistic simulations of the higher-temperature operating mode for the Proposed Action inventory, nominal scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations	I-77

Figure | Page

I-13 Total annual individual dose at the RMEI location for 300 probabilistic simulations of the lower-temperature operating mode for the Proposed Action inventory, nominal scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations I-77

I-14 Total annual individual dose at the RMEI location for 300 probabilistic simulations of the higher-temperature operating mode for the Module 1 inventory, nominal scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations I-78

I-15 Total annual individual dose at the RMEI location for 300 probabilistic simulations of the higher-temperature operating mode for the Module 2 incremental inventory, nominal scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations I-78

I-16 Comparison plot of the mean total annual individual dose at the RMEI location for the higher-temperature operating mode for the Proposed Action, Module 1, and incremental Module 2 inventories, nominal scenario I-79

I-17 Total annual individual dose at the RMEI location for 500 out of 5,000 probabilistic simulations of the higher-temperature operating mode for the Proposed Action inventory under the igneous activity scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of all 5,000 simulations I-79

I-18 Total mean individual receptor dose at the RMEI location for the higher-temperature operating mode for the Proposed Action inventory under the igneous activity scenario; the figure displays the mean results for both the eruptive and intrusive events and the sum of these events as "Total Igneous" I-80

I-19 Total mean annual individual dose at the RMEI location for the higher-temperature and lower-temperature operating modes for the Proposed Action inventory under the igneous activity scenario I-80

I-20 Total annual individual dose at the RMEI location for 500 out of 5,000 probabilistic simulations of the lower-temperature operating mode for the Proposed Action inventory under the igneous activity scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations I-81

I-21 Total mean annual individual dose at the RMEI location for the lower-temperature operating mode for the Proposed Action inventory under the igneous activity scenario; the figure displays the mean results for both the eruptive and intrusive events and the sum of these events as "Total Igneous" I-81

I-22 Total annual individual dose at the RMEI location for 300 probabilistic simulations of the higher-temperature operating mode for the Proposed Action inventory under the human-intrusion-at-30,000-years scenario; the figure also displays the median, mean, and 95th-percentile values of these simulations I-82

I-23 Mean activity concentrations of gross alpha activity and total radium (radium-226 plus radium-228) at the RMEI location of 300 probabilistic simulations of the higher-temperature operating mode for the Proposed Action inventory for the nominal scenario I-82

I-24 Mean activity concentrations of gross alpha activity and total radium (radium-226 plus radium-228) at the RMEI location of 300 probabilistic simulations of the lower-temperature operating mode for the Proposed Action inventory for the nominal scenario I-83

Figure Page

I-25 Mean dose to critical organs for technetium-99, carbon-14, and iodine-129 at the RMEI location of 300 probabilistic simulations of the higher-temperature operating mode for the Proposed Action inventory for the nominal scenario I-83

I-26 Mean dose to critical organs for technetium-99, carbon-14, and iodine-129 at the RMEI location of 300 probabilistic simulations of the lower-temperature operating mode for the Proposed Action inventory for the nominal scenario I-84

I-27 Fraction of perforated cladding for commercial spent nuclear fuel as a function of time after repository closure I-84

I-28 Release rate of carbon-14 from the repository to the ground surface for 80,000 years following repository closure I-85

APPENDIX I. ENVIRONMENTAL CONSEQUENCES OF LONG-TERM REPOSITORY PERFORMANCE

This appendix provides detailed supporting information on the calculation of the environmental consequences of long-term repository performance (postclosure, up to 1 million years). Chapter 5 summarizes these consequences for the Proposed Action, and Chapter 8, Section 8.3.1 summarizes the cumulative impacts of Inventory Modules 1 and 2.

Section I.1 introduces the bases for analysis of long-term performance. Section I.2 provides an overview of the use of computational models developed for the Total System Performance Assessment (TSPA) model, that was used for the analysis of long-term impacts to groundwater in this environmental impact statement (EIS). Section I.3 identifies and quantifies the inventory of waste constituents of concern for analysis of long-term performance. Section I.4 details the modeling extensions to the TSPA *nominal scenario* [Proposed Action inventory, reasonably maximally exposed individual (RMEI) location at approximately 18 kilometers, or 11 miles, downgradient of the potential repository, and no disruptive events other than seismic] developed to estimate potential impacts for expanded inventories. An estimate of how the impacts might change for locations beyond the RMEI location is also provided. Section I.5 provides detailed results for waterborne radioactive material impacts, while Section I.6 provides the same for waterborne chemically toxic material impacts. Section I.7 describes atmospheric radioactive material impacts. To aid readability, all the figures are placed at the end of the appendix.

HOW ARE THE TOTAL SYSTEM PERFORMANCE ASSESSMENT MODEL AND THIS EIS ANALYSIS RELATED?

The analysis of long-term performance for this EIS builds incrementally on the TSPA model.

This appendix is primarily concerned with those aspects of the EIS analysis of long-term performance that are incremental over the TSPA model. Only those parts of the analysis unique to this EIS are detailed in this appendix, and the text refers to the appropriate TSPA model documents for information on the bases of the analyses. Some aspects of the modeling detailed in the TSPA are repeated in this appendix in overview form to provide continuity and enhance understanding of the approach.

For a full understanding of all details of the analysis of long-term performance in this EIS, it is necessary to study not only this appendix but also the other TSPA model documents cited herein.

I.1 Introduction

This EIS analysis of postclosure impacts used and extended the modeling work performed for the Yucca Mountain site suitability evaluation that supports the site recommendation process. The EIS analysis relied on the GoldSim program computer simulation model (DIRS 151202-Golder Associates 2000, all) used by DOE to calculate radiological doses resulting from waterborne releases through the groundwater pathway.

Analysis of long-term performance for this EIS required several steps. The EIS analysis model started with the TSPA model, which was modified as discussed below. For this EIS the modeling (described in this appendix) was further expanded to evaluate the impacts for expanded waste inventories (see Section I.4). Additional calculations provided estimates of how the impacts would vary for two other distances [30 kilometers (19 miles) downgradient, and the discharge location that is 60 kilometers (37 miles) downgradient at Franklin Lake Playa (refer to Section I.4.5)], analysis of long-term groundwater impacts of chemically toxic materials, and estimates of atmospheric radiological doses to the local population.

The model used to evaluate long-term impacts of radioactive materials in the groundwater simulates the release and transport of radionuclides away from the repository into the unsaturated zone, through the unsaturated zone, and ultimately through the saturated zone to the accessible environment. Analysis of long-term performance depends greatly on the underlying process models necessary to provide thermal-hydrologic conditions, near-field geochemical conditions, unsaturated zone flow fields, and saturated zone flow fields as a function of time. Using these underlying process models involves multiple steps that must be performed sequentially before modeling of the overall system can begin.

Figure I-1 shows the general flow of information between data sources, process models, and the TSPA model. Several process-level computer models are identified in Figure I-1. Examples are the site- and drift-scale thermal hydrology model and the saturated zone flow and transport model. The process models are very large and complex computer software programs used in detailed studies to provide information to the TSPA model. These process models are generally where fundamental laboratory and field data are introduced into the modeling. The subsystem and abstracted models section of the figure encompasses those portions of the TSPA model that are modeled within the GoldSim program. Examples are the unsaturated zone flow fields and the biosphere dose conversion factors. These models are generally much simpler than the process models. They are constructed to represent the results of the more detailed process modeling studies. Often they are simple functions or tables of numbers. This is the process referred to as *abstraction*. It is necessary for some of these subsystem models to be quite complex, even extensive computer codes. The ultimate result sought from modeling long-term performance is a characterization of radiological dose to humans with respect to time, shown at the top of the TSPA section of the figure. This is accomplished by assessing behavior at intermediate points and "handing" off the results to the next subsystem in the primary release path.

ABSTRACTION

Abstraction is the distillation of the essential components of a process model into a suitable form for use in a TSPA. The distillation must retain the basic intrinsic form of the process model but does not usually require its original complexity. Model abstraction is usually necessary to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and uncertainty analyses.

I.2 Total System Performance Assessment Methods and Models

DOE conducted analyses for this EIS to evaluate potential long-term impacts to human health from the release of radioactive materials from the Yucca Mountain Repository. The analyses were conducted in parallel with, but distinct from, the TSPA calculations for the site suitability evaluation. The methodologies and assumptions are detailed in the *Total System Performance Assessment for the Site Recommendation* (DIRS 153246-CRWMS M&O 2000, all), and the *FY01 Supplemental Science and Performance Analyses* (DIRS 155950-CRWMS M&O 2001, all). These two versions of the model are referred to respectively here as the "Site Recommendation model" and the "Supplemental Science and Performance Analyses model." Note that the Supplemental Science and Performance Analyses model starts with the Site Recommendation model and includes incremental enhancements to several parts of the Site Recommendation model. Further changes were made to the model to meet distinct requirements of this EIS. These changes are discussed in more detail in Section I.4 and in DIRS 157307-BSC (2001, Enclosure 1). In summary, the changes are as follows:

- The biosphere dose conversion factors are based on the Reasonably Maximally Exposed Individual (RMEI) defined in 40 CFR 197.21.

- The length of the saturated zone simulated in the performance-assessment model extends from the edge of the repository to where the principal flow path crosses north latitude 36 degrees 40 minutes 13.6661 seconds, as the point where the RMEI would reside. This location is approximately 2 kilometers (1.2 miles) north of the intersection of U.S. Route 95 and Nevada State Route 373, a location formerly known as "Lathrop Wells" and currently known as "Amargosa Valley," that is approximately 20 kilometers (12 miles) downgradient from the repository.

- The groundwater protection standard using an annual water usage of 3.7 million cubic meters per year (exactly 3,000 acre-feet per year) was used in calculating the gross-alpha activity, the total radium concentration, and the total organ dose. All other concentrations were calculated using the same water usage as the Site Recommendation and the Supplemental Science and Performance Analyses models.

- The analysis used the waste inventory that was presented in *Inventory Abstraction* (DIRS 154841-BSC 2001, all). The difference between this inventory and that used in the Site Recommendation and Supplemental Science and Performance Analyses models is that for analysis purposes, U.S. Navy spent nuclear fuel is conservatively modeled as commercial spent nuclear fuel (DIRS 152059-BSC 2001, all and DIRS 153849-DOE 2001, Section 4.2.6.3.9, p. 4-257) and not as DOE-owned spent nuclear fuel.

- Waste package corrosion for the calculations in this report was due to general corrosion independent of temperature.

- The process-level lower-temperature repository operating mode thermal-hydrologic results were corrected to include radiation heat transfer.

- The model was expanded to accommodate inventories other than that for the Proposed Action.

The TSPA is a comprehensive systems analysis in which models of appropriate levels of complexity represent all important features, events, and processes to predict the behavior of the system being analyzed and to compare this behavior to specified performance standards. In the case of the proposed Yucca Mountain Repository system, a TSPA must capture all of the important components of both the engineered and the natural barriers. In addition, the Yucca Mountain TSPA must evaluate the overall uncertainty in the prediction of waste containment and isolation, and the risks caused by the uncertainty in the individual component models and corresponding parameters.

The components of the Yucca Mountain Repository system include five major elements that the TSPA must evaluate for the nominal scenario:

- The natural environment unperturbed by the presence of underground openings or emplaced wastes

- Perturbations to the natural system caused by construction of the underground facilities, waste emplacement, and expected natural events (such as seismic behavior)

- The long-term degradation of the engineered components designed to contain the radioactive wastes

- The release of the radionuclides from the engineered containment system

- The migration of these radionuclides through the engineered and natural barriers to the biosphere and their potential uptake by people, leading to a radiation dose consequence

The analysis included models associated with such disruptive events as volcanism and human intrusion (drilling). Sections I.2.10 and I.2.11 provide an overview of the processes and the models used to represent these disruptive events.

The EIS analysis of long-term performance represents a "snapshot in time," and ongoing work will help refine that snapshot. In the meantime, DOE believes the results of this EIS analysis are conservative estimates, and that work currently in progress or planned will increase confidence in the overall modeling approach.

The calculations for the TSPA model and calculations for this EIS were performed within a probabilistic framework combining the most likely ranges of behavior for the various component models, processes, and related parameters. In some cases, bounding conservative values were used where the available data did not support development of a realistic range. This appendix presents the results as time histories of annual radiological dose to an individual over 10,000 and 1 million years following repository closure. As noted above, the TSPA model implements some of the individual process models directly, while other process models run outside the TSPA model to produce *abstractions* in the form of data tables, response surfaces, or unit-response functions. The TSPA model provides a framework for incorporating these abstractions and integrating them with other subsystem models. This is done in a *Monte Carlo* simulation-based methodology to create multiple random combinations of the likely ranges of the parameter values related to the process models. Probabilistic performance of the entire waste-disposal system was computed in terms of radiological dose to individuals at selected distances from the repository.

MONTE CARLO METHOD: UNCERTAINTY

An analytical method that uses random sampling of parameter values available for input into numerical models as a means of approximating the uncertainty in the process being modeled. A Monte Carlo simulation comprises many individual runs of the complete calculation using different values for the parameters of interest as sampled from a probability distribution. A different outcome for each individual calculation and each individual run of the calculation is called a *realization* (DIRS 153246-CRWMS M&O 2000, p. A-55).

The methodology for analysis of long-term performance for this EIS draws on the extensive analyses performed in support of the TSPA model. Most of the process models (and their abstractions) developed for the TSPA model were used directly in the analyses described in this appendix. Components that were modified to account for the additional analyses considered in this EIS are emphasized in this appendix. However, for continuity, the sections that follow include a general overview of all the elements of the TSPA model.

I.2.1 FEATURES, EVENTS, AND PROCESSES

The first step in the TSPA is to decide which representations of possible future states of the proposed repository (scenario classes and scenarios) are sufficiently important to warrant quantitative analysis. The TSPA model can analyze only a relatively small number of the essentially infinite combinations of features, events, and processes that could affect the system. It is important, therefore, that the scenarios chosen for analysis provide a sound basis for evaluating the performance of the repository. Specifically, the chosen scenarios must be representative of the conditions of greatest relevance to forecasting the long-term behavior of the system.

The first step in developing scenarios is to make an exhaustive list of features, events, and processes that could apply to the repository system. The initial list is developed using a number of resources:

- Lists previously compiled by other organizations on an international scale (such as the Nuclear Energy Agency of the Organization for Economic Cooperation and Development)
- Lists compiled during earlier stages of site exploration
- Lists developed by experts from the Yucca Mountain Project and outside consultants

The starting list is subjected to a comprehensive screening process. Features, events, and processes are screened from the list based on several criteria:

- Obvious inapplicability to the specific site (for example, the starting list included processes that occur only in salt, a rock type known to be not present at Yucca Mountain).
- Very low probability of occurrence (for example, meteorite impact)
- Very low consequence to the closed repository (for example, an airplane crash)
- Exclusion by regulatory direction (for example, deliberate human intrusion)

FEATURES, EVENTS, AND PROCESSES

Features are physical parts of the system important to how the system could perform. Examples include the Ghost Dance Fault and the Topopah Spring stratigraphic unit.

Events are occurrences in time that can affect the performance or behavior of the system. Events tend to happen in short periods in comparison to the period of concern, and they tend to occur at unpredictable times. Examples include a volcanic intrusion or a human intrusion by drilling.

Processes are physical and chemical changes that occur over long periods, tend to be 100-percent likely to occur, and are predictable. Examples include corrosion of the metals in the waste package and dissolution of waste form materials after exposure to water.

The remaining features, events, and processes are combined in scenarios that incorporate sequences of events and processes in the presence of features. The three main scenarios evaluated are:

- Nominal scenario (generally undisturbed performance with only seismic events)
- Volcanism scenario (eruption through the repository or intrusion of igneous material into the repository)
- Inadvertent human intrusion scenario.

When the scenarios described above were formed from the Features, Events, and Processes retained after screening, the focus was on the 10,000-year compliance period. Therefore in the screening documentation the reliance on a limit of 10,000-years was sometimes expressed. This EIS is charged by 40 CFR Part 197 with the task of reporting the peak dose values whenever they occur during the period of geologic stability. As can be seen by the results in this EIS, the peaks occur at times considerably longer than 10,000 years and it was necessary to carry out the analysis for 1 million years in order to establish the peak dose. Because the TSPA model used to generate all the results in the EIS is the same model that resulted from the Features, Events, and Processes screening it is important to explore the possible effect of the use of a 10,000 limit when screening Features, Events, and Processes. The following discussions are provided by the Features, Events, and Processes screening staff for that purpose (DIRS 155937-Freeze 2001, all). In addition to the discussions from the DIRS 155937-Freeze (2001, all) document there is also a short discussion of seismic Features, Events, and Processes. For a comprehensive discussion of all the Features, Events, and Processes the reader is referred to the Features, Events, and Processes database documentation (DIRS 154365-Freeze, Brodsky, and Swift 2001, all).

Note that in numbers given in the headings or text of Sections I.2.1.1 through I.2.1.7 (in the form "FEP No. X.X.X.X.X") refer to an index number from the Features, Events, and Processes database (DIRS 154365-Freeze, Brodsky, and Swift 2001, all).

I.2.1.1 Tectonic Activity (FEP No. 1.2.01.01.00)

The current strain rate is indicated by DIRS 118952-Savage, Svarc, and Prescott (1999, p. 17627) as less than 2 millimeters per year (0.08 inch per year) and is reflected in local slip rates of between 0.001 and 0.03 millimeters per year (0.0004 and 0.001 inches per year). At the highest rate, the total slip after 10,000 years would be on the order of 0.010 to 0.3 meters (0.03 to 1 foot), but after 1 million years could be on the order of 1 to 30 meters (3.3 to 98 feet). The increased rates of tectonic and igneous activity in the geologic past (and leading to the 30-meter value) were associated with greater crustal strain rates than exist currently. In particular, DIRS 118942-Fridrich (1999, all) indicate extension of the Crater Flat structural basin to have been on the order of 18 to 40 percent between about 12.6 and 11.6 million years ago during the major pulse of extension, with the rate of extension declining exponentially since 11.6 million years ago. From the late Quaternary through the present, the rate of extension is less than 1 percent of the initial rate. These studies suggest that crustal extension rates are likely to vary insignificantly or to decrease with time. As a consequence, assumption of the existing tectonic setting and strain rates for periods out to 1 million years, for purposes of the EIS, is reasonable, although quantification of associated displacements would exhibit a time-dependent increase in uncertainty.

The median probability for exceeding fault displacements greater than 3 meters (10 feet) on the Solitario Canyon Fault is approximately 0.0001 in 10,000 years, and the median and mean probability for fault displacement on intrablock faults of 2 meters (6.6 feet) or greater is less than 0.0001 in 10,000 years (DIRS 100354-USGS 1998, all). The projected values assume that the tectonic strain rate is either equal to or less than the existing strain rate. Projection and use of these displacements for a 1-million-year time frame is appropriate, but is accompanied with an increase in uncertainty in the probable displacement value.

Based on the repository design, the drifts could accommodate as much as 2 meters (6.6 feet) of vertical displacement on intrablock faults before waste package shearing conditions could occur and, with the use of set-backs, at least 3 meters (10 feet) of offset could be accommodated in the Solitario Canyon Fault, and possibly more if distributed faulting is considered. Hypothetical models at the mountain-scale also suggest that flow in fault zones and fractures would not be significantly affected by displacement of as much as 10 meters (33 feet). The tolerance values are not time-dependent. The projected total slip values at 1 million years (1 to 30 meters, or 3.3 to 98 feet) are of the same order of magnitude as the tolerance limit (1 to 10 meters, or 3.3 to 33 feet).

Because the tolerance values are the same order of magnitude as the projected total slip, and because the tectonic setting and history of the site suggest that strain rates will either vary insignificantly or decrease, the assumptions and models in the TSPA related to tectonic activity should be reasonable and applicable for the 1-million-year time span as well.

I.2.1.2 Erosion/Denudation (FEP No. 1.2.07.01.00)

Erosion is a process that is expected to be ongoing at Yucca Mountain. The maximum erosion over 10,000 years is expected to be less than 10 centimeters (3.9 inches) (DIRS 100520-YMP 1993, p. 55), which is within the range of existing surface irregularities.

After 1 million years the maximum total erosion would be 10 meters (33 feet), assuming the erosion rate estimated for the next 10,000 years remained constant for the next 1 million years. This maximum value is far less than the amount required to expose waste at the land surface, and possible effects would therefore be limited to changes in infiltration and flow in the unsaturated zone. Local changes of as much as 10 meters would represent a small change relative to the hundreds of meters (thousands of feet) of topographic variability already incorporated in the infiltration model used to calculate flow in the unsaturated zone. The effects of erosion on infiltration are therefore considered negligible. Erosion due to normal surface processes at Yucca Mountain is therefore excluded from the 1 million-year analyses.

Future climate projections extending to 10,000 years (DIRS 136368-USGS 2000, all; DIRS 153038-CRWMS 2000, all) indicate that, although the climate is expected to evolve to a cooler, wetter climate, conditions will be that of a glacial transition or glacial-type climate. As a result, direct glacial erosion and transport is not considered a credible event. Therefore, glacial erosion is excluded on the basis of low probability.

The effects of erosion processes on how radionuclides might accumulate in soils and subsequently enter the biosphere are included (DIRS 136281-CRWMS M&O 2000, Section 6.1.1) for the post-10,000-year period. The effects of erosional processes in the biosphere are considered in an Analysis Model Report titled *Evaluate Soil/Radionuclide Removal by Erosion and Leaching* (DIRS 136281-CRWMS M&O 2000, all) and are considered in *Total System Performance Assessment for the Site Recommendation* (DIRS 153246-CRWMS M&O 2000, Sections 3.9 and 3.10) as part of the peak dose calculations.

I.2.1.3 Periglacial Effects (FEP No. 1.3.04.00.00)

This process refers to climate conditions that could produce a cold, but glacier-free, environment. Results of such a climate could include permafrost (permanently frozen ground). Some consequences of such a condition identified in the secondary Features, Events, and Processes are enhanced erosion due to the freeze/thaw cycle and the trapping of gases in or near the proposed repository.

Global climate change was addressed in the TSPA using a climate model based on paleoclimate information. That is, the record of climate changes in the past was used to predict changes in climate for the future. Because the geologic record indicates that climatic conditions during the Quaternary period (the past 1.6 million years) at no time resulted in plant communities at Yucca Mountain that are consistent with periglacial conditions (DIRS 136281-CRWMS M&O 2000, Section 4.2.4), this process has been excluded on the basis of low probability.

Future climates are described in terms of discrete climate states that are used to approximate continuous variations in climate. The effects of seasonality are included in the climate model by using climate analogs with specific seasonal meteorological records. More specific information about the methods used to predict future climate change and the findings for the climate model is provided in DIRS 136368-USGS (2000, Section 6). Climate modeling is incorporated in the TSPA through the unsaturated zone flow fields, which have different surface-water infiltration as a result of different climates. A description of the modeling methods used for infiltration and how infiltration is affected by climate is in DIRS 136368-USGS (2000, Section 6).

Potential future climate conditions at Yucca Mountain were analyzed in two Analysis Model Reports: *Future Climate Analyses* (DIRS 136368-USGS 2000, all) and *Documentation of Million-Year TSPA* (DIRS 153038-CRWMS 2000, all). The climate at Yucca Mountain for the next 10,000 years is treated as a sequence of three climate states: modern (interglacial) climate for 400 to 600 years, monsoon climate for 900 to 1,400 years, and glacial-transition (intermediate) climate for the balance of the 10,000-year period. The glacial-transition (intermediate) climate occurs either preceding or following the colder, wetter full glacial climate states. Three additional full-glacial climate states are specified during the longer period of 1 million years, with different climate stages synchronized with the earth orbital clock. Full-glacial stages would encompass about 21 percent of the time over the next 1 million years. The intermediate climate would be the dominant climate for the next 1 million years.

I.2.1.4 Glacial and Ice Sheet Effects (FEP No. 1.3.05.00.00)

This process refers to the local effects of glaciers and ice sheets. Paleoclimate records indicate that glaciers and ice sheets have not occurred at Yucca Mountain at any time in the past (DIRS 136368-USGS 2000, Section 6.2). The closest alpine glaciers to Yucca Mountain during the Pleistocene were in the Sierra Nevada of California and possibly the Spring Mountains in Nevada (DIRS 151945-CRWMS M&O 2000, Section 4.2.3.3.6), too far from Yucca Mountain to have any effect on site geomorphology or hydrology. Given the relatively low elevation of Yucca Mountain, there is no credible mechanism by which a glacier could form at the site over the next 10,000 years, and there is no evidence to suggest formation at Yucca Mountain in the next 1 million years. Therefore, this process is excluded on the basis of low probability. Note, however, that the regional climatic effects of ice sheets that might form farther north are included based on a change in climate states.

I.2.1.5 Hydrostatic Pressure on Container (FEP No. 2.1.07.04.00)

A repository at Yucca Mountain would emplace waste above the water table in a fractured, porous medium. Thus, the pressure on the waste package is approximately atmospheric under present conditions. Possible changes in the elevation of the water table due to climate change and tectonic processes have been evaluated (DIRS 153931-CRWMS M&O 2001, Sections 6.2.11 and 6.2.8; DIRS 154826-BSC 2001, Section 6.7.6), and water table fluctuations due to climate change are included in the TSPA model. Even under the wettest future climate states, however, the highest elevation of the water table would be far below the emplacement drifts, and hydrostatic pressure effects on the packages are therefore excluded on the basis of low probability for both 10,000-year and 1-million-year analyses.

I.2.1.6 Soil and Sediment Transport (FEP No. 2.3.02.03.00)

Transport of soil and sediments in the biosphere is discussed in the Analysis Model Reports titled *Evaluate Soil/Radionuclide Removal by Erosion and Leaching* (DIRS 136281-CRWMS M&O 2000, all) and *Nominal Performance Biosphere Dose Conversion Factor Analysis* (DIRS 152539-CRWMS M&O 2001, all). The results of these analyses are used in Sections 3.9 and 3.10 of the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000). Aeolian and fluvial transport of contaminated volcanic ash has been indirectly included in the TSPA–Site Recommendation igneous disruption scenario through the use of a wind direction fixed toward the critical group for all hypothetical eruptions. As described in Section 3.10 of TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000), use of a fixed wind direction compensates for the lack of an explicit model for sediment transport following ash deposition by ensuring that all eruptions would result in the deposition of contaminated ash at the location of the critical group, regardless of the wind direction at the time of the event. The TSPA–Site Recommendation calculations include the probability of eruptive events extending past the 10,000-year regulatory period to calculate peak dose.

Paleoclimate records indicate that glaciers and ice sheets have not occurred at Yucca Mountain at any time in the past (DIRS 136368-USGS 2000, Section 6.2). The closest alpine glaciers to Yucca Mountain during the Pleistocene were in the Sierra Nevada of California and possibly the Spring Mountains in Nevada (DIRS 151945-CRWMS M&O 2000, Section 4.2.3.3.6), too far from Yucca Mountain to have any effect on site geomorphology or hydrology. Given the relatively low elevation of Yucca Mountain, there is no credible mechanism by which a glacier could form at the site within the time frames considered. Therefore, glacial transport of soil and sediments is not considered credible and this process is excluded on the basis of low probability.

I.2.1.7 Seismic Damage to Waste Packages

This discussion refers to the following Features, Events, and Processes:

- Seismic vibration causes container failure (FEP No. 1.2.03.02.00)
- Mechanical impact on waste container and drip shield (FEP No. 2.1.03.07.00)
- Effects and degradation of drip shield (FEP No. 2.1.06.06.00)
- Rockfall – large block (FEP No. 2.1.07.01.00)
- Mechanical degradation or collapse of drift (FEP No. 2.1.07.02.00)

These events all have to do with possible damage to the waste packages or drip shields either directly or indirectly (for example, rock fall) due to seismic events. In the Features, Events, and Processes screening these events were screened out for low consequence because up to 10,000 years the waste packages remain essentially intact (see detailed results in Section I.5) and possess their original design strength. Because the packages are designed to withstand seismic events that are of sufficient likelihood during the 10,000-year period, it follows that a low-consequence screening for the 10,000-year period is justified.

The analysis for the million-year period extended the screening of seismic damage to waste packages throughout that time. This was an analytical assumption based on using the best data and models available for the Final EIS. No quantitative analysis was performed to determine when a waste package might degrade to the point where it could be damaged by a seismic event. However, it is reasonable to expect that peak dose estimates would likely have been higher (by an unknown amount) if the analysis accounted for potential seismic damage of degraded waste packages hundreds of thousands of years into the future.

I.2.2 UNSATURATED ZONE FLOW

Changes in climate over time provide a range of conditions that determine how much water could fall onto and infiltrate the ground surface. Based on current scientific understanding, the current climate is estimated to be the driest that the Yucca Mountain vicinity will ever experience. All future climates were assumed similar to current conditions or wetter than current conditions. The *climate* model provides a forecast of future climates based on information about past patterns of climates (DIRS 153246-CRWMS M&O 2000, p. 3-38 to 3-42). This is generally accepted as a valid approach because climate is known to be cyclical and largely dependent on repeating patterns of earth orbit and spin. The model represents future climate shifts as a series of instant changes. During the first 10,000 years, there are three changes, in order of increasing wetness, from present-day to a monsoon and then to a glacial-transition climate. Between 10,000 years and 1 million years there are 45 changes between six climate states incorporated in the TSPA model (DIRS 153246-CRWMS M&O 2000, p. 3-38):

CLIMATE CHANGE

The analysis of long-term performance considered six climate states. Many changes in climate states occur in the simulation over a 1-million-year period after closure. The times of change are keyed to known past cycles for the previous million years as determined by paleoclimatology studies. (DIRS 153246-CRWMS M&O 2000, Figure 3.2-16, p. F3-24).

- Interglacial Climate (same as present day)
- Intermediate Climate (same as the Glacial-transition)
- Intermediate/Monsoon Climate
- Three stages of Glacial Climate of varying infiltration rates

Precipitation that is not returned to the atmosphere by evaporation or transpiration enters the unsaturated zone flow system. Water infiltration is affected by a number of factors related to climate, such as an increase or decrease in vegetation on the ground surface, total precipitation, air temperature, and runoff. The *infiltration* model uses data collected from studies of surface infiltration in the Yucca Mountain region (DIRS 155950-BSC 2001, Section 3.2.2). It treats infiltration as variable in the region, with more occurring along the crest of Yucca Mountain than along its base. The results of the climate model affect assumed infiltration rates. For each climate, there is a set of three infiltration rates (high, medium, low) and associated probabilities. This forms a discrete distribution that is sampled in the probabilistic modeling. The sampled ranges are described in Tables I-1 and I-2. Whenever a particular climate state is in effect, the associated infiltration rate distribution is sampled for each realization of the simulation.

Table I-1. Average net infiltration rates (millimeters per year) over the unsaturated zone flow and transport model domain for the present-day, monsoon, and glacial transition climate states.[a,b]

Climate	Lower bound	Mean	Upper bound
Present day	1.3	4.6	11.1
Monsoon	4.6	12.2	19.8
Glacial transition	2.5	17.8	33.0

a. Adapted from DIRS 155950-BSC (2001, Table 3.3.2-1).
b. To convert from millimeter per year to inch per year, multiply by 0.03937.

Table I-2. Average net infiltration rates (millimeters per year) over the unsaturated zone flow and transport model domain for full-glacial climate states.[a,b]

Climate	Lower bound	Mean	Upper bound
Glacial, Stage 8/10	33.0[c] (36.0[d])	87.9	151.0
Glacial, Stage 6/16	24.4	87.9	151.0
Glacial, Stage 4	12.9	24.4	87.9

a. Adapted from DIRS 155950-BSC (2001, Table 3.3.2-3).
b. To convert from millimeter per year to inch per year, multiply by 0.03937.
c. Derived using upper-bound intermediate climate meteorological station data (DIRS 155950-BSC 2001, Tables 3.3.1-5, 3.3.1-6).
d. Derived using alternate Stage 6/16 meteorological station data (DIRS 155950-BSC 2001, Tables 3.3.1-5, 3.3.1-6).

Water generally moves downward in the rock matrix and in rock fractures. The rock mass at Yucca Mountain is composed of volcanic rock that is fractured to varying degrees because of contraction during cooling of the original, nearly molten rock and because of extensive faulting in the area. Water flowing in the fractures moves much more rapidly than water moving through the matrix. At some locations, water might collect in locally saturated zones (*perched water*) or might be laterally diverted because of differing rock properties at rock layer interfaces. The overall unsaturated flow system is heterogeneous, and the locations of flow paths, velocities, and volumes of groundwater flowing along these paths are likely to change many times over the life of the repository system. The *mountain-scale unsaturated zone flow* model assumes constant flow over a specific period (taken from the infiltration model) and generates three-dimensional flow fields for three different infiltration boundary conditions, the six different climates described above, and several values of rock properties (DIRS 153246-CRWMS M&O 2000, pp. 3-29 and 3-41). The model is an isothermal model; thermal effects can be neglected because flow would be strongly perturbed only by heat near the emplacement drifts and at early times (DIRS 153246-CRWMS M&O 2000, p. 3-31). The influence of heat near the drifts is dealt with in the thermal hydrology models discussed below. The flow fields from the mountain-scale unsaturated zone flow model are the abstractions that are utilized by the TSPA model while the system model is running. The TSPA model simply switches to the correct flow field for the sampled infiltration rate, as dictated by the current climate state and sampling of the infiltration rate range.

After water returns to the repository walls, it would drip into the repository. The number of seeps that would occur and the amount of water that would be available to drip would be restricted by the low rate at which water flows through Yucca Mountain, which is in a semiarid area. Drips would occur only if the hydrologic properties of the rock mass caused the water to concentrate enough to feed a seep. Over time, the number and locations of seeps would increase or decrease, corresponding to increased or decreased infiltration based on changing climate conditions. The *seepage flow* model calculates the amount of seepage that could occur based on input from the unsaturated zone flow model (DIRS 155950-BSC 2001, Section 4.3). The basic conceptual model for seepage suggests that openings in unsaturated rock act as capillary barriers and divert water around them. For seepage to occur in the conceptual model, the rock pores at the drift wall would have to be locally saturated. Drift walls could become locally saturated by either disturbance to the flow field caused by the drift opening or variability in the permeability field that created channeled flow and local ponding. Of the two reasons, the variability effect is more important. Drift-scale flow calculations made with uniform hydrologic properties suggest that seepage would not occur at expected percolation fluxes. However, calculations that include permeability variations do estimate seepage, with the amount depending on the hydrologic properties and the incoming percolation flux. The seepage abstraction is based on extensive modeling calibrated by measurements from onsite testing in the Exploratory Studies Facility (DIRS 153246-CRWMS M&O 2000, pp. 3-35 to 3-36, and DIRS 155950-BSC 2001, Section 4.3.1.5). The seepage abstraction includes probability distributions for the fraction of waste packages encountering seepage and the seep flow rate, accounting for parameter uncertainty, spatial variability, and other effects, such as focusing (DIRS 155950-BSC 2001, Section 4.3.2), episodicity (DIRS 155950-BSC 2001, Section 4.3.5), rock bolts (DIRS 155950-BSC 2001, Section 4.3.3), drift degradation (DIRS 155950-BSC 2001, Section 4.3.4) and coupled processes (DIRS 155950-BSC 2001, Sections 4.3.5 through 4.3.7). All of these parameters are input as uncertainty distributions that are sampled in the probabilistic modeling.

I.2.3 ENGINEERED BARRIER SYSTEM ENVIRONMENTS

Engineered barrier system environments refer to the thermodynamic and chemical environments in the emplacement drifts. These environments control processes that affect the components of the engineered barrier system (such as the drip shields, waste packages, and waste forms). The environmental characteristics of importance are the degradation of the drift (including rock fall into the drift), temperature, relative humidity, liquid saturation, pH, liquid composition, and gas composition. Thermal effects on flow and chemistry outside the drifts are also important because they affect the amount and composition of water and gas entering the drifts. The engineered barrier system environments are important to long-term repository performance because they would help determine degradation rates of components, quantities and species of mobilized radionuclides, transport of radionuclides through the drift into the unsaturated zone, and movement of fluids into the unsaturated zone.

The *drift degradation model* describes the deterioration of the rock mass surrounding the repository emplacement drifts. Deterioration would occur by failure of fractures that bound blocks of rock at the drift walls and the resultant falling of those blocks into the drift. The deterioration is described in terms of key block analysis (DIRS 153246-CRWMS M&O 2000, pp. 3-43), which is a tool used for the following purposes:

- Provide a statistical description of block sizes formed by fractures around the emplacement drifts
- Estimate changes in drift profiles due to fallen blocks of rock
- Provide an estimate of the time required for significant drift deterioration to occur.

Key blocks would be formed by the intersection of three or more fracture planes with the excavation. Key blocks could become dislodged and fall because of seismic effects. A detailed analysis, based on observation and testing, was used to develop an abstraction of block failures and rockfalls. The abstraction is in the form of tables of numbers and volumes of blocks falling per unit length of emplacement drift as a function of time due to seismic and other effects.

Within the TSPA model, most engineered system calculations were performed for a limited number of waste package locations. In the model, each of these locations is representative of a group of waste packages with similar environmental characteristics. Radionuclide releases, for example, were calculated for a representative waste package and then scaled up by the number of failed waste packages in the group. Not all waste packages in a group would fail at the same time because additional variability is included in the waste package degradation calculation. The waste package groups (referred to as *bins*) are not based on physical location. Rather, the bins are based on infiltration patterns (that is, divided into categories of specific ranges of infiltration rate) and on waste type (that is, codisposal packages and commercial spent nuclear fuel packages) (DIRS 153246-CRWMS M&O 2000, Section 3.3.2).

The heat generated by the decay of nuclear materials in the repository would cause the temperature of the surrounding rock and waste packages to rise from the time of emplacement until a few hundred years after repository closure (DIRS 153246-CRWMS M&O 2000, Figure 3.3-9, p. F3-33). The water and gas in the heated rock would be driven away from the repository during this period, referred to in this EIS as the *thermal pulse*. The thermal output of the materials would decrease with time; eventually, the rock would return to its original temperature, and the water and gas would flow back toward the repository. The *multi-scale thermal hydrology model* is used to study the processes that would govern the temperature, relative humidity, liquid saturation, liquid flow rate, liquid evaporation rate, and thermal effects on seepage. Drift-scale modeling includes coupling of drift-scale processes with mountain-scale processes to account for effects such as faster cooling of waste packages near the edge of the repository, as compared to waste packages near the center. A multi-scale modeling and abstraction method was developed to couple drift-scale processes with mountain-scale processes (DIRS 153246-CRWMS M&O 2000, pp. 3-56 to 3-58, and DIRS 155950-BSC 2001, Section 5.3.1). In addition, a coupled *thermal-hydrology-chemistry model* was developed to study the coupled effects on the heat, flow and chemistry of the system (DIRS 153246-CRWMS M&O 2000, p. F3-33). The results of these detailed modeling studies are abstracted as response surfaces of temperature, humidity, and liquid saturation.

The source term for transport of radionuclides from the proposed repository in the unsaturated zone and saturated zone water flow is the radionuclide flux from inside the drifts to the unsaturated zone rock. That flux would be influenced by the in-drift engineered barrier system chemical environment. The *engineered barrier system geochemical environment models* (DIRS 153246-CRWMS M&O 2000, pp. 3-62 to 3-69 and DIRS 155950-BSC 2001, Sections 6.3.1 and 6.3.3) were used to study the changing composition of gas, water, colloids, and solids in the emplacement drifts under the perturbed conditions of the repository. Several submodels were integrated to provide detailed results and interpretations. The major composition changes would be caused by the thermal loading of the system and the emplacement of large masses of materials that can react with water and gas in the system. The system would continually change due to the heating and cooling cycle. Because the emplaced materials would be very different from the host rock, the entering water and gas would be altered by reaction with these materials. Emplaced materials could be an additional source of colloids that could affect how radionuclides were transported in the aqueous system. The engineered barrier system geochemical environment models produce detailed results that are then abstracted for the following processes:

- Water and cement interactions
- Gas and water interactions
- Evaporation of water and condensation of vapor
- Salts precipitation and dissolution
- Microbial activity and effects
- Corrosion and degradation of engineered barrier system components

- Water and invert interactions
- Water and colloids interactions.

The abstractions were integrated into the TSPA model as chemistry lookup tables for various periods, parametric results, and sometimes enhancement or correction factors for other processes such as corrosion or transport (DIRS 153246-CRWMS M&O 2000, pp. 3-69 to 3-79 and DIRS 155950-BSC 2001, Sections 5.3.2.2 and 6.3.1.6).

The location of the seeps would depend to some extent on the natural conditions of the rock but also on the alterations caused by the construction of a repository. Alterations, such as increased fracturing, would be caused by mechanical processes related to drilling the drifts or by thermal heating and expansion of the drift walls. The alterations in the seepage could also be caused by chemical alterations occurring as the engineered materials dissolved in water and reprecipitated in the surrounding rock, closing the pores and fractures. The chemistry in the drift would change continually because of the complex interactions between the incoming water, circulating gas, and materials in the drift (for example, concrete from the liner or metals in the waste package). The changes in chemistry would be strongly influenced by heat during the thermal pulse.

The seepage would flow through the engineered barrier system along eight pathways. These pathways are (DIRS 155950-BSC 2001, Sections 8.2 and 8.3):

1. Seepage flux entering the drift—This would be the liquid flow into the engineered barrier system.

2. Flow through the drip shield—Liquid flux through the drip shield would begin after holes formed due to general corrosion.

3. Diversion around the drip shield—The portion of the flux that did not flow through the drip shield was assumed to bypass the invert and flow directly into the unsaturated zone.

4. Flow through the waste package—The fluid flow through the waste package would be based on the presence of holes due to general corrosion. The liquid flux through any holes in the waste package is calculated using a flux splitting algorithm that incorporates the fraction of the waste package or drip shield that has openings. This algorithm considers the projected patch area on a breached waste package or drip shield.

5. Flow diversion around the waste package—The portion of the flux that did not flow through the drip shield and onto the waste package was assumed to bypass the waste form and flow directly onto the invert.

6. Evaporation from the invert condensation underneath the drip shield—The magnitude of the evaporative flux from the invert would be based on the thermal-hydrologic abstraction.

7. Flow from the waste package to the invert—All flux from the waste package would flow to the invert, independent of breach location on the waste package. The presence of the emplacement pallet was ignored, and the waste package was assumed to be lying on the invert so a continuous liquid pathway for diffusive transport would exist at all times.

8. Flow through the invert into the unsaturated zone—Flow could be by advection or diffusion. The model accounts for sorption in the invert.

The model accounts for the evaporation of some of the liquid flux to the drip shield (DIRS 155950-BSC 2001, Section 8.3.1.3). The evaporation rate at the top of the drip shield would be bounded by the amount of heat available to vaporize water on the upper portion of the drip shield. This heat flow rate into the upper portion of the drip shield was used to determine the maximum volumetric flow rate of incoming seepage water that could be completely vaporized at this location.

I.2.4 WASTE PACKAGE AND DRIP SHIELD DEGRADATION

The radioactive waste placed in the proposed repository would be enclosed in a two-layer waste package. The layers would be of two different materials that would fail at different rates and from different mechanisms as they were exposed to various repository conditions. The outer layer would be a high-nickel alloy metal (Alloy-22) and the inner layer a stainless-steel alloy metal (316NG). To divert dripping water away from the waste package and thereby extend waste package life, a Titanium Grade 7 drip shield would be placed over the waste packages just prior to repository closure. The drip shield would divert water entering the drift from above preventing seep water from contacting the waste package. The *drip shield and waste package degradation models* were used to simulate the degradation of these components (DIRS 153246-CRWMS M&O 2000, pp. 3-79 to 3-91, DIRS 155950-BSC 2001, Section 7, and DIRS 157307-BSC 2001, Enclosure 1). Three main types of degradation were considered in the nominal scenario: humid-air general corrosion, aqueous general corrosion, and stress corrosion cracking. Two additional corrosion processes—microbially induced corrosion and thermal aging/phase instability—were considered to provide enhanced general corrosion on the waste package. General corrosion mechanisms would be conceptually similar for the drip shield and waste package, and were simulated using a common approach. Mechanical failure by rockfall was screened out of the model due to low consequence.

The primary models supplying input to the drip shield and waste package degradation abstractions are the thermal hydrology model and the in-drift geochemical abstraction model. Output from the degradation models is a time-dependent quantitative assessment of the drip shield and waste package degradation and failure. Results include the time to initial breach for the drip shield and the waste package; time to first breach of the waste package by stress corrosion crack failure; and the degree of drip shield and waste package failure as a function of time. The time of the first breach of the waste package would correspond to the start of waste form degradation in the breached package. The output also includes the uncertainty and spatial variation of the degradation information for each waste package and drip shield at different locations (described above as *bins*) within the potential repository. A recent reevaluation of potential early waste package failure mechanisms indicated that improper heat treatment of waste packages could lead to a gross failure of affected waste packages, although the probability of this occurrence is very low. Therefore, improper heat treatment of waste packages is now modeled in the current waste package degradation analysis (DIRS 155950-BSC 2001, Section 7.3.6). An analysis of manufacturing and testing led to a probability distribution for the number of packages that could fail from improper heat treatment of the Alloy-22 closure weld. The resulting distribution is listed in Table I-3. The distribution for waste package failures reflects a very conservative view, because it is assumed that if the outer weld was not properly heat treated the package would automatically fail, even though improper heat treatment would not necessarily result in failure, and the inner weld on the Alloy-22 and the inner stainless steel weld would probably remain intact. This distribution was sampled for each realization of the TSPA model and resulted in early failures of a very small number of waste packages in some of the realizations. This would result in very small releases during the first 10,000 years after closure.

The analysis in this EIS assessed the possible effects of waterborne chemically toxic materials. The analysis did not identify any organic materials as being present in sufficient quantities to be toxic. A screening process eliminated most other materials because they were not of concern for human health effects (see Section I.6.1). Some of the components of the high-nickel alloy (such as chromium, molybdenum, nickel, and vanadium) would be of sufficient quantity and possible toxicity to warrant

Table I-3. Poisson probabilities for improper heat treatment of waste packages.[a]

	Proposed Action		Inventory Module 1		Inventory Module 2	
Number of packages	Probability	Cumulative probability	Probability	Cumulative probability	Probability	Cumulative probability
0	0.76874	0.76874	0.69011	0.69011	0.98669	0.98669
1	0.20218	0.97092	0.25596	0.94608	0.013224	0.999911
2	0.026587	0.99751	0.047468	0.99354	8.8615×10^{-5}	0.9999996
3	2.3308×10^{-3}	0.99984	5.8687×10^{-3}	0.99941	3.9588×10^{-7}	1
4	1.5325×10^{-4}	0.999992	5.4417×10^{-4}	0.999957	1.32464×10^{-9}	1
5	8.0608×10^{-6}	1	4.0367×10^{-5}	0.9999974	3.5555×10^{-12}	1

a. Calculated from the mean Poisson value entered in the performance model.

further analysis. The rate of release of these materials was taken directly from data used for the waste package degradation modeling.

I.2.5 WASTE FORM DEGRADATION

The *waste form degradation model* evaluates the interrelationship among the in-package water chemistry, the degradation of the waste form (including cladding), and the mobilization of radionuclides (DIRS 153246-CRWMS M&O 2000, pp. 3-92 to 3-129 and DIRS 155950-BSC 2001, Sections 9.3.1-9.3.2, 10.3.1, and 10.3.4). The model consists of components that:

- Define the radioisotope inventories for representative commercial spent nuclear fuel and codisposal waste packages (this is the inventory abstraction discussed in more detail in Section I.3.1)

- Evaluate in-package water chemistry—in-package chemistry component abstraction (using chemistry lookup tables developed from detailed process model studies and calculations involving other model parameters)

- Evaluate the matrix degradation rates for commercial spent nuclear fuel, DOE-owned spent nuclear fuel, and high-level radioactive waste forms—waste form matrix degradation component abstractions (a temperature- and pH-dependent rate equation with several parameters, such as rate constants and activation energies, represented by statistical distributions)

- Evaluate the rate of Zircaloy cladding degradation (in the case of commercial spent nuclear fuel)—cladding degradation component abstraction with the following components:

  – Initial failure of Zircaloy cladding represented by a triangular distribution (low, mode, and high fraction of rods failed)

  – Creep failure of Zircaloy cladding represented by a series of triangular distributions, with a low value, mode value and high value, for fraction of rods perforated; each distribution for a specific peak waste package temperature range

  – Localized corrosion of Zircaloy cladding represented as a function of the water flux into the waste package, or a small, constant rate if there is no seepage

  – Assumption of total perforation of all stainless-steel cladding at time zero

  – Seismically induced cladding failure as all cladding would fail when a discrete event frequency of 0.0000011 per year occurred

- A cumulative distribution of cladding unzipping rate coefficients; the coefficients are multiplied by the fuel matrix dissolution rate to obtain unzipping velocity
- Effective exposure area of matrix (for radionuclide distribution) as a function of cladding perforation and unzipping

- Evaluate the radionuclide concentrations for aqueous phases—dissolved radionuclide concentration component abstraction (distributions of solubilities as a function of pH and temperature in the waste package; solubilities are also checked for possible limitations due to waste form degradation rate or package inventory)
- Evaluate diffusion of radionuclides in the waste package (DIRS 155950-BSC 2001, Section 10.3.1)
- Evaluate sorption of radionuclides in the waste package (DIRS 155950-BSC 2001, Section 10.3.4)
- Evaluate the waste form colloidal phases—colloidal radionuclide concentration component abstraction (reversible and irreversible colloid models)

I.2.6 ENGINEERED BARRIER TRANSPORT

The waste form would be the source of all radionuclides considered for the engineered barrier system. Radionuclides could be transported downward through the invert and into the unsaturated zone. Transport could occur by diffusion or by advection, depending on the route of the transport. The *engineered barrier system transport abstraction* (DIRS 153246-CRWMS M&O 2000, pp. 3-130 to 3-143) conservatively assumes that diffusion could occur once stress corrosion cracks form, regardless of whether conditions were appropriate for a continuous liquid pathway to exist. Colloid-facilitated transport of radionuclides was included as a transport mechanism. Radionuclides would be transported from the waste package either as dissolved species or bound in, or attached to, colloids.

The abstraction simulates the following transport modes:

- Waste package to invert path
  - Diffusion through stress corrosion cracks
  - Diffusion and advection through patches failed by bulk corrosion

- Invert to unsaturated zone path - Diffusion, sorption and advection through the invert (DIRS 155950-BSC 2001, Section 10.3.3 and 10.3.4)

Diffusion is represented by a diffusion transport equation with an empirical effective diffusivity that is a function of liquid saturation, porosity, and temperature. Sorption on corrosion products is characterized by a linear isotherm (K_D). Advective transport is represented by a liquid transport equation with the velocity determined by the engineered barrier system flow abstraction discussed above.

I.2.7 UNSATURATED ZONE TRANSPORT

Unsaturated zone transport refers to the movement of radionuclides from the engineered barrier system of the proposed repository, through the unsaturated zone, and to the water table. The unsaturated zone would be the first natural barrier to radionuclides that escaped from the potential repository. The unsaturated zone would act as a barrier by delaying radionuclide movement. If the delay was long enough for significant decay of a specific radionuclide, the unsaturated zone could have a significant effect on the ultimate dose from releases of that radionuclide to the environment. The *unsaturated zone transport model* (DIRS 153246-CRWMS M&O 2000, pp. 3-144 to 3-156, and DIRS 155950-BSC 2001, Section 11.3) is used to describe how radionuclides move through the unsaturated zone. The unsaturated zone model considers transport through welded tuff and nonwelded tuff and flow through both the fractures and the rock matrix. In addition, the model accounts for the existence of zeolitic alterations in some regions. These zeolitic tuffs are characterized with low permeability and enhanced radionuclide sorption.

The unsaturated zone water flow would provide the background on which the unsaturated zone transport took place. The model uses the flow fields developed using the unsaturated zone flow model, as described in Section I.2.2. Radionuclides can migrate in groundwater as dissolved molecular species or by being associated with colloids. Five basic processes affect the movement of dissolved or colloidal radionuclides:

- Advection (movement of dissolved and colloidal material with the bulk flow of water) including drift shadow effects on the seepage below the repository (DIRS 155950-BSC 2001, Section 11.3.1)
- Diffusion (movement of dissolved or colloidal material because of random motion at the molecular or colloidal particle scale)
- Sorption (a combination of chemical interactions between solid and liquid phases that reversibly partition radionuclides between the phases)
- Hydrodynamic dispersion (spreading of radionuclides perpendicular to and along the path of flow as they transport caused by localized variations in the flow field and by diffusion)
- Radioactive decay

Sorption is potentially important because it slows, or retards, the transport of radionuclides. Diffusion of radionuclides out of fractures into matrix pores is also a potential retardation mechanism because matrix transport is generally slower than fracture transport. However, sorption and matrix diffusion have less effect on colloids, so radionuclides bound to colloids can be more mobile than radionuclides dissolved in water. Radioactive decay could be important both from quantity reduction of certain radionuclides and the behavior of decay products that can have different transport properties than the decayed radionuclide.

The unsaturated zone transport model was implemented in the TSPA model as an embedded computer code that simulates the three-dimensional transport using a residence-time, transfer-function, particle-tracking technique. The key parameters such as sorption coefficients, diffusion coefficients, dispersivity, fracture spacing, and colloid parameters (partitioning, retardation, colloid size, fraction of colloids exchanging between matrix units) are all input as uncertainty distributions. The results are expressed as breakthrough curves (normalized fraction of total amount of radionuclide arriving at the saturated zone as a function of time) for each radionuclide. These are the inputs for saturated zone transport modeling.

I.2.8 SATURATED ZONE FLOW AND TRANSPORT

The saturated zone at Yucca Mountain is the region beneath the ground surface where rock pores and fractures are fully saturated with groundwater. The upper boundary of the saturated zone is called the water table. The proposed repository would be approximately 300 meters (1,000 feet) above the water table in the unsaturated zone.

As on the surface, underground water flows down the hydraulic gradient. Based on water-level observations in area wells, groundwater near Yucca Mountain flows generally in a north-to-south direction. The major purpose of the *saturated zone flow and transport model* (DIRS 153246-CRWMS M&O 2000, pp. 3-156 to 3-174, and DIRS 155950-BSC 2001, Sections 12.3.1 and 12.3.2) is to evaluate the migration of radionuclides from their introduction at the water table below the potential repository to the point of release to the biosphere (for example, a water supply well). Radionuclides can move through the saturated zone either as a dissolved solute or associated with colloids. The input to the saturated zone is the spatial and temporal distribution of mass flux of radionuclides from the unsaturated zone. The output of the saturated zone flow and transport model is a mass flux of radionuclides in the water used by a hypothetical farming community.

I.2.8.1 Saturated Zone Flow

The *saturated zone flow submodel* (DIRS 153246-CRWMS M&O 2000, pp. 3-157 to 3-164 and DIRS 155950-BSC 2001, Section 12.3.1) takes inputs from the unsaturated zone flow submodel and produces outputs, in the form of flow fields, for the saturated zone transport submodel. The saturated zone flow submodel incorporates a significant amount of geologic and hydrologic data taken from drill holes near Yucca Mountain. The saturated groundwater flow in the vicinity of Yucca Mountain can be estimated by knowing the porosity of the flow media, the hydraulic conductivity, and the recharge of water into the flow media. The primary tool used to describe saturated zone flow is a numerical model formulated in three dimensions. The three-dimensional saturated zone flow model has been developed specifically to determine the groundwater flow field at Yucca Mountain. The model was used to produce a library of flow fields (maps of groundwater fluxes). In addition, a GoldSim-based one-dimensional version of the model was used to provide flow information for a one-dimensional model of transport of radionuclide decay products.

I.2.8.2 Saturated Zone Transport

The *saturated zone transport submodel* (DIRS 153246-CRWMS M&O 2000, pp. 3-157 to 3-164, and DIRS 155950-BSC 2001, Section 12.3.2) takes inputs in the form of radionuclide mass fluxes from the unsaturated zone transport submodel and produces outputs in the form of radionuclide mass fluxes to the biosphere model. The saturated zone transport model incorporates a substantial amount of laboratory and field data taken from a variety of sources.

Radionuclides released from a repository at Yucca Mountain to the groundwater would enter the saturated zone beneath the repository and would be transported first southeast, then south, toward the Amargosa Desert. The radionuclides could be transported by the groundwater in two forms: as dissolved species or associated with colloids. Dissolved species typically consist of radionuclide ions complexed with various groundwater species, but still at molecular size. Colloids are particles of solids, typically clays, silica fragments, or organics, such as humic acids or bacteria, that are larger than molecular size, but small enough to remain suspended in groundwater for indefinite periods. Colloids are usually considered to have a size range of between a nanometer and a micrometer. A radionuclide associated with a colloid can transport either attached to the surface or bound within the structure of the colloid.

Transport through the saturated zone was primarily modeled using a three-dimensional particle-tracking method (DIRS 153246-CRWMS M&O 2000, pp. 3-168 to 3-169). The three-dimensional transport model was not used directly by the TSPA model. It was used to generate a library of breakthrough curves—distributions of transport times that are used, along with a time-varying source term from the unsaturated zone, to calculate the releases at the geosphere/biosphere boundary. The model accounts for the flow of groundwater and its interaction with varying media along the flow path. In the volcanic rocks that comprise the saturated media in the immediate vicinity of Yucca Mountain, groundwater flows primarily through fractures, while a large volume of water is held relatively immobile in the surrounding rock matrix. Radionuclides would travel with the moving fracture water but, if dissolved, could diffuse between the matrix water and fracture water. This transfer between fracture and matrix water is characteristic of a dual-porosity system. The saturated zone transport model is a dual-porosity model.

The media at greater distances from Yucca Mountain are alluvial gravels, sands and silts. The model simulates these areas as a more uniform porous material. While there is a possibility for channelized flow in the alluvium, current data indicate little evidence of dual-porosity behavior that would indicate this (DIRS 155950-BSC 2001, p. 12-23).

A one-dimensional saturated zone transport model was used to account for decay and ingrowth during transport. This model was incorporated directly in the GoldSim model as a series of pipes. The advantage of using the one-dimensional model is that the radionuclide masses can be accounted for directly. The disadvantage is that the flow and transport geometry is necessarily simplified.

I.2.9 BIOSPHERE

If the radionuclides were removed from the saturated zone in water pumped from wells, the radioactive material could result in dose to humans in several ways. For example, the well water could be used to irrigate crops that would be consumed by humans or livestock, to water stock animals that would be consumed by humans as dairy or meat products, or to provide drinking water for humans. In addition, if the water pumped from irrigation wells evaporated on the ground surface, the radionuclides could be left as fine particulate matter that could be picked up by the wind and inhaled by humans. The *biosphere pathway model* (DIRS 153246-CRWMS M&O 2000, pp. 3-175 to 3-187) was used to predict radiation exposure to a person living in the general vicinity of the repository if there was a release of radioactive material to the biosphere after closure of the proposed repository. The model uses a biosphere dose conversion factor that converts saturated zone radionuclide concentrations to annual individual radiation dose. The biosphere dose conversion factor was developed by analyzing the multiple pathways through the biosphere by which radionuclides can affect a person. The biosphere scenario assumed a *reference person* living in the Amargosa Valley region at various distances from the repository. People living in the Town of Amargosa Valley would be the group most likely to be affected by radioactive releases. An adult who lives year-round at this location, uses a well as the primary water source, and otherwise has habits (such as the consumption of local foods) similar to those of the inhabitants of the region. Because changes in human activities over millennia are unpredictable, the analysis assumed that the present-day reference person described future inhabitants. Strict definitions for the reference person (the Reasonably Maximally Exposed Individual, or RMEI) have been prescribed in 40 CFR Part 197. The chemically toxic materials were not evaluated in the biosphere model because there are no usable comparison values for radiologic and nonradiologic dose. Rather, a separate analysis of concentrations of these materials was made. The concentrations were then compared to available regulatory standards, such as the Maximum Contaminant Level Goal if available, or to the appropriate Oral Reference Dose.

The biosphere is the last component in the chain of TSPA model subsystem components. There are two connections between the biosphere submodel and other TSPA model submodels. One is for the groundwater irrigation scenario (nominal scenario), where the biosphere is coupled to the saturated zone flow and transport model; and the other is for the disruptive scenario, where the biosphere is coupled to the volcanic dispersal model. For the human intrusion scenario, the biosphere model is coupled with the saturated zone flow and transport model, and the event is treated as a perturbation to the nominal scenario. The groundwater path doses are based on specific paths of groundwater flow derived from regional data.

The primary result of the biosphere modeling is the construction of biosphere dose conversion factor distributions for the groundwater-release scenarios and the volcanic-ash-release scenario (DIRS 157307-BSC 2001, all). For the nominal scenario, well withdrawal of groundwater is the source of water for drinking, irrigation, and other uses. A farming community at the point of withdrawal would use the water at a rate based on surveys of current usage. The hypothetical farming community consists of between 15 and 25 farms supporting about 100 people. All radionuclides reaching this community in groundwater were assumed to be mixed in the volume of water that the community would use (this is the concept of full "capture" of the total plume of contamination). The water usage was input as a distribution of values based on current water usage data. The exposure pathways routes taken by radionuclides through the biosphere from the source to an individual are typical for a farming community in this environment. Farming activities usually involve more exposure pathways than other human activities in the Yucca Mountain region, including ingestion of contaminated water and locally produced food as well as inhalation and direct exposure from soil contamination intensified by the significant outdoor activity inherent in a farming lifestyle.

During periods of very wet climate, the Amargosa Desert is actually a lake and the irrigated farm scenario on which the biosphere model is based is not applicable. This is consistent with regulatory guidance that indicates no attempt should be made to project future human behavior and lifestyles (even if driven by climate change). The approach used is conservative because the use of groundwater for irrigation and domestic purposes has the effect of bringing up relatively concentrated solutions of contaminants. In a scenario where the Amargosa Desert is a lake (as it was 20,000 years ago), this large quantity of water would dilute the radionuclides to very low concentrations. Furthermore, the use of water would follow a greatly altered pattern. Consideration of all this leads to the conclusion that peak doses would be much lower than those projected in the current analysis.

I.2.10 VOLCANISM

Igneous activity (flow of volcanic material as in a volcanic eruption) has been identified as a disruptive event that has a potential to affect repository long-term performance. Yucca Mountain is in a region that has had repeated volcanic activity in the geologic past. Although the probability of recurrence at Yucca Mountain during the next 10,000 years is small, it is greater than 1 chance in 10,000 and is, therefore, retained as a scenario.

If igneous activity occurred at Yucca Mountain, possible effects on the repository could be grouped into three areas:

- Igneous activity that would not directly intersect the repository (can be shown to have no effect on dose from the repository)

- Volcanic eruptions in the repository that would result in waste material being entrained in the volcanic magma or pyroclastic material, bringing waste to the surface (resulting in atmospheric transport of volcanic ash contaminated with radionuclides and subsequent human exposure downwind)

- An igneous intrusion intersecting the repository (no eruption but damage to waste packages from exposure to the igneous material that would enhance release to the groundwater and, thus, enhance transport to the biosphere)

Based on studies of past activity in the region, probabilities for different types of igneous activity were estimated. Each type of event was described in detail based on observation of effects of past activities. These descriptions include geometry of intrusions, geometry of eruptions, physical and chemical properties of volcanic materials, eruption properties (velocity, power, duration, volume, and particle characteristics). Most of the parameters describing the igneous activity were entered in the modeling as probability distributions.

A collection of different modeling approaches was used to develop responses to the different types of activity described above (DIRS 153246-CRWMS M&O 2000, pp. 3-187 to 3-216 and DIRS 157307-BSC 2001, Enclosure 1).

I.2.11 HUMAN INTRUSION

Human intrusion was modeled based on a stylized scenario that is a conceptualization of the assumptions outlined in the Environmental Protection Agency standard (DIRS 157307-BSC 2001, Enclosure 1). The assumptions are based on recommendations of the National Research Council of the National Academy of Sciences. The Council observed that it is not possible to predict human behavior over the extremely long periods of concern and prescribed the scenario as a reasonable representation of typical inadvertent intrusion.

The models used were the same as those for the nominal scenario, except a source term was introduced for the time of the intrusion. This source term is characteristic of direct penetration of a waste package with a drill bit (DIRS 157307-BSC 2001, Enclosure 1).

I.2.12 NUCLEAR CRITICALITY

A nuclear criticality occurs when sufficient quantities of fissionable materials come together in a precise manner and the required conditions exist to start and sustain a nuclear chain reaction. One of the required conditions is the presence of a moderator, such as water, in the waste package. The waste packages would be designed to make the probability of a criticality occurring inside the waste package extremely small. In addition, based on an analysis of anticipated repository conditions, it is very unlikely that a sufficient quantity of fissionable materials could accumulate outside the waste packages in the precise configuration and with the required conditions to create a criticality. If, somehow, an external criticality was to occur, analyses indicate that it would have only minor effects on repository performance. In the unlikely event that a criticality occurred, there would be a short-duration localized rise in temperature and pressure, as well as an insignificant increase in the repository radionuclide inventory. No measurable effect on repository performance would result from this event (DIRS 153849-DOE 2001, p. 4-416).

I.2.13 ATMOSPHERIC RADIOLOGICAL CONSEQUENCES

In addition to the groundwater pathway, the analysis of long-term performance evaluated potential consequences of the release of radioactive gases into the environment. An analysis separate from the groundwater modeling described in the previous sections was used to forecast such consequences. The model used results from the waste package degradation models to evaluate when waste packages and fuel cladding would fail and, therefore, release contained radioactive gases. This model provided input to release and transport estimates for the atmospheric pathway. Section I.7 contains details of this analysis.

I.3 Inventory

This section discusses the inventories of waterborne radioactive materials used to model radiological impacts and of some nonradioactive, chemically toxic waterborne materials used in the repository environment that could present health hazards. This section also discusses the inventory of atmospheric radioactive materials.

I.3.1 INVENTORY FOR WATERBORNE RADIOACTIVE MATERIALS

There would be more than 200 radionuclides in the materials placed in the repository (see Appendix A of this EIS). In the Proposed Action, these radionuclides would be present in five basic waste forms: commercial spent nuclear fuel, mixed-oxide fuel and plutonium ceramic (called here *plutonium disposition waste*), borosilicate glass formed from liquid wastes on various DOE sites known as high-level radioactive waste, DOE-owned spent nuclear fuel, and naval spent nuclear fuel (DIRS 153246-CRWMS M&O 2000, Figure 3.5-4). In the repository, these wastes would be placed in several different types of waste packages of essentially the same construction but of varying sizes and with varying types of internal details. (DIRS 150558-CRWMS M&O 2000, Section 4.3). It is neither necessary nor practical to model the exact configuration of waste packages. The individual details of each package design are not significant parameters in modeling the processes involved in waste package degradation, waste form degradation, and radionuclide transport from the engineered barrier system. Constructing a TSPA model with each individual package and its unique design would result in a computer model too large to run on any available computer in a practical time. Therefore two representative types of waste packages were modeled in representative zones of the repository. The development of the two representative types of waste packages and their radionuclide inventories is the process of abstraction.

An abstracted inventory was used in the analysis of long-term groundwater impacts in much the same way as many other Features, Processes and Events were abstracted. The TSPA model is a high-level system model that performs hundreds of trials in a Monte Carlo framework. To make such a calculation tractable, it was necessary to reduce highly complex descriptions or behaviors to simplified concepts that capture the essential characteristics. In the case of inventory, the highly complex array of waste streams for the five fundamental waste categories (commercial spent nuclear fuel, plutonium-disposition waste, high-level radioactive waste, DOE-owned spent nuclear fuel, and naval spent nuclear fuel) were considered in developing the abstraction to representative waste packages that capture the essential features of the total inventory of radionuclide materials. The waste packages in the repository can be represented in two package types: a commercial spent nuclear fuel waste package and a codisposal waste package containing DOE spent nuclear fuel and high-level radioactive waste glass. The naval spent nuclear fuel was modeled as part of the commercial spent nuclear fuel. The plutonium disposition waste was split into the commercial spent nuclear fuel packages (mixed-oxide fuel) and codisposal package (immobilized plutonium within a high-level radioactive waste container) (DIRS 154841-BSC 2001, all).

The abstracted inventory has been carefully developed to maintain essential characteristics of the waste forms for the purpose of input to the TSPA model. As such, the TSPA abstracted inventory cannot be used for any other purpose, because it is not reality but rather a representation of reality that works only for the purpose intended. The averaging, blending, and screening of radionuclides to reduce the total number, while retaining essential physical characteristics of the waste, were all tailored to the TSPA model. Therefore, any attempt to compare this abstracted inventory with other abstractions used for other analyses in the repository will not be valid. The only essential comparison that can be made is that of the fundamental inputs to the abstraction process to fundamental inputs used in other analyses.

The abstraction of the inventory is shown in Figure I-2. In the figure, items in boxes are references to documents that either describe an analysis or are a data transmittal. The items not in boxes (next to the arrows) are the data produced from a documented analysis and used in another documented analysis.

Figure I-2 identifies four fundamental inputs:

- Input from DOE Environmental Management's National Spent Nuclear Fuel Program that identifies the characteristics of all DOE-owned spent nuclear fuel that would be sent to the repository (DIRS 110431-INEEL 1999, all)

- A body of high-level radioactive waste data collected from the EIS Data Call of 1997 (DIRS 104418-Rowland 1997); this includes information concerning high-level radioactive waste and plutonium-disposition waste

- A body of data that forms the database for commercial spent nuclear fuel; this is a collection of documents including key documents such as DOE/RW-0184 (DIRS 102588-DOE 1992, all) in its various revisions

- The *Monitored Geologic Repository Project Description Document* (DIRS 151853-CRWMS M&O 2000, all).

These four inputs were manipulated in various analyses that were brought together in the inventory abstraction (DIRS 157307-BSC, 2001, all) and are shown as the box at the bottom center of the figure. The fundamental data on commercial spent nuclear fuel was first processed in three analyses: simulation of a delivery schedule to the repository (DIRS 119348-CRWMS M&O 1999, all) (this was done using a standard computer code called CALVIN and source term studies for boiling-water reactor and pressurized-water reactor fuel that describe the typical radionuclide inventories for these spent fuels (DIRS 136428-CRWMS M&O 1999, all) (DIRS 136429-CRWMS M&O 1999, all). The CALVIN results are part of the input to the source term studies. All of the commercial spent nuclear fuel studies were then combined in a packaging study that describes the resulting spent nuclear fuel packages in the detailed design of the repository (DIRS 138239-CRWMS M&O 2000, all). The fundamental information on high-level radioactive waste was analyzed to determine decay and ingrowth of radionuclides in the waste and obtain inventory as a function of time (DIRS 147072-CRWMS M&O 1999, all). This study used the ORIGEN-S computer code, a standard code for determining inventories as a function of time. Fundamental data on DOE-owned spent nuclear fuel was analyzed to determine a packaging strategy (DIRS 149005-CRWMS M&O 2000, all). The results of this study identified three canister types and their inventories. At this point the results of commercial spent nuclear fuel, high-level radioactive waste, and DOE-owned spent nuclear fuel analyses were brought together in another analysis to develop a set of 13 standard package configurations (DIRS 153909-BSC 2001, all). This result was the basic set of detailed package configurations for the repository.

Another important analysis is the screening analysis. In this analysis, the contribution of specific radionuclides to inhalation and ingestion dose was determined and the radionuclides were ranked according to their contribution to total dose of all radionuclides (DIRS 153597-CRWMS M&O 2000, all). The metric for screening radionuclides is the radiation dose that a radionuclide could impose on a human living in the vicinity of Yucca Mountain. Identification of the important dose contributors is based on an estimate of the amount of radionuclides that could reach a human (DIRS 136383-CRWMS M&O 2000, all). Identification of the important dose contributors involves three steps:

1. For the waste form under consideration, the relative dose contribution from an individual radionuclide is calculated by multiplying its inventory abundance (in terms of its radioactivity) by its dose conversion factor (a number that converts an amount of a radionuclide into the dose that a human would incur if the radionuclide was ingested, inhaled, or came in close proximity). This multiplication gives a result that can be compared to values derived in the same manner for other radionuclides to determine the more important contributors to the dose.

2. The individual radionuclides are ranked, with the highest contributor to the dose given the highest ranking, and the percent contribution of each radionuclide in the list to the total dose (the sum of the doses from the radionuclides in the list) is calculated.

3. Radionuclides that are included in the analysis are the highest-ranked radionuclides that, when their dose contributions are combined, produce 95 percent of the dose.

These steps identify which radionuclides would be included in the dose estimate, if all the radionuclides in a waste form were released to the environment in proportion to their inventory abundance. However, radionuclides are not always released in proportion to their inventory abundance. Factors that can affect releases of radionuclides, depending on the scenario being considered, include radionuclide longevity, solubility, and transport affinity.

Radionuclide longevity is the lifetime of a radionuclide before it decays. Solubility is the amount of a radionuclide that will dissolve in a given amount of water. Transport affinity is a radionuclide's potential for movement through the environment. This movement can involve a number of mechanisms, for example: fracture flow (the movement of radionuclides with water flowing in fractures), matrix diffusion (the diffusion of radionuclides from water in the fractures into water in the matrix), or colloidal-facilitated transport (the movement of radionuclides associated with small particles of rock or waste form degradation products). Transport affinity is not a measurable property, but a qualitative description of the likelihood of transport. If a group of radionuclides is transported via a particular mechanism, and that mechanism dominates release, the group of radionuclides will be preferentially released (relative to radionuclides not in the group) to the environment. If a radionuclide has a short half-life, it will have a higher activity in the waste form at early times (close to repository closure); however, at later times, the radionuclide will have all but disappeared from the waste form. If a radionuclide is not soluble in the near-field environment around the waste package, it may not be released to the environment through groundwater transport, even if it is abundant and available.

Because radionuclide longevity, solubility, and transport affinity can affect releases of radionuclides, the identification of important dose contributors includes examination of possible "what-if" scenarios that could result in releases of radionuclides to the environment. For example, "What if radionuclide releases are the result of a colloidal transport mechanism? If the steps described above are applied to the subset of radionuclides that could be released through a colloidal-transport mechanism (radionuclides that readily bind to rock and colloidal particles), which of those radionuclides would be identified as the important contributors to dose?" Or, "What if a volcanic direct release to the environment occurs? If the steps described above are applied to the radionuclides present in the waste form in a direct release, which of those radionuclides would be identified as the important contributors to dose?" The radionuclide screening examined over 1,200 potential what-if scenarios and identified the important dose contributors for each one. The cases examined consider times from 100 to 1 million years after repository closure (100, 200, 300, 400, 500, 1,000, 2,000, 5,000, 10,000, 100,000, 300,000, and 1,000,000 years); eight waste forms (average and bounding pressurized-water reactor fuel, average and bounding boiling-water reactor fuel, average and bounding DOE spent nuclear fuel, and average and bounding DOE high-level radioactive waste); three transport affinity groups (highly sorbing, moderately sorbing, and slightly to nonsorbing); and two exposure pathways (inhalation and ingestion).

In addition to the radionuclides selected based on contribution to dose, other radionuclides (in particular radium-226 and radium-228) must be considered because of the groundwater protection requirements in 40 CFR Part 197. Other radionuclides must also be considered in the analysis because they belong to decay chains; they must be included to accurately track other members of the decay chains. (A decay chain is a sequence of radionuclides that, because of radioactive decay, change from one to the other; thus, the amount of one is dependent on the amounts of the others.)

The complete list of radionuclides produced by the screening merges all the lists of radionuclides developed from the various scenarios. For example, if a radionuclide is important for estimating the dose from DOE spent nuclear fuel, it is included in the analysis even though these waste forms would occupy a small fraction of the repository. Similarly, if a radionuclide is important for estimating the dose from the highly sorbing transport group, it is included in the analysis, even if analyses show that colloid transport is a minimal contributor to release.

The inventory abstraction then took as input the 13 configurations, the design of the repository, the screening analysis, and a special americium-241 ingrowth analysis (DIRS 153596-CRWMS M&O 2001, all). The abstraction provided two fundamental results:

- The total initial inventory for the TSPA model for the Proposed Action represented as the quantity of radionuclides in two representative waste package types. The total number of radionuclides represented has been reduced by a screening process with two criteria: elimination of all radionuclides with a half-life less than 20 years and inclusion of all radionuclides that contribute at least 95 percent of the total radiological dose.

- A recommended list of radionuclides to track for each of three scenarios: nominal scenario, disruptive events (volcanism) scenario, and human intrusion scenario. Not all radionuclides in the master list are necessarily included in a particular scenario. This is because some radionuclides are not important in some scenarios.

Additional analyses for this EIS included consideration of two other inventories that are not part of the Proposed Action. These analyses supported the analysis of cumulative impacts from possible future actions. The first of these is the addition of more commercial spent nuclear fuel. The combined inventory of the Proposed Action and this additional commercial spent nuclear fuel is referred to as Inventory Module 1. In addition, a category for Greater-Than-Class-C plus Special-Performance-Assessment-Required materials (Inventory Module 2 only) could be added in the future. The waste packages in this calculation include the commercial spent nuclear fuel packages and DOE spent nuclear fuel and high-level radioactive waste codisposal packages described in DIRS 150558-CRWMS M&O (2000, all). This EIS assumes that the Inventory Module 2 Greater-Than-Class-C and Special-Performance-Assessment-Required waste would be packaged in codisposal waste packages (DIRS 155393-CRWMS M&O 2000, Attachment II). The numbers of idealized waste packages used in the calculations are listed in Table I-4.

Table I-4. Modeled number of idealized waste packages by category type for the abstracted inventories of the Proposed Action, Inventory Module 1, and Inventory Module 2.

Waste category	Proposed Action	Inventory Module 1	Inventory Module 2
Commercial spent nuclear fuel[a]	7,860	11,754	0[b]
DOE spent nuclear fuel/high-level radioactive waste codisposal	3,910	4,877	0[b]
Greater-Than-Class-C	0	0	201
Special-Performance-Assessment-Required	0	0	400
Total	***11,770***	***16,631***	***601[b]***

a. 300 U.S. Navy spent nuclear fuel waste packages are modeled as commercial spent nuclear fuel waste packages.
b. Inventory Module 2 would include all packages in Inventory Module 1 plus the numbers shown for Greater-Than-Class-C and Special-Performance-Assessment Required waste packages; however, for modeling purposes only the *incremental increase* in the number of waste packages was modeled and the result added to the result for Inventory Module 1 impacts to estimate Inventory Module 2 impacts.

The physical properties of the various waste forms to be placed in the proposed Yucca Mountain repository are described in detail in DIRS 151109-CRWMS M&O (2000, all).

I.3.1.1 Radionuclide Inventory Used in the Model of Long-Term Performance for Proposed Action

The tabulated per-waste-package inventory used in the Proposed Action calculations is listed in Table I-5.

I.3.1.2 Radionuclide Inventory Used in the Model of Long-Term Performance for Inventory Module 1

The abstracted per-waste-package radionuclide inventory used for the Proposed Action also applies to additional waste packages for the expansion of the repository to include all potential commercial and DOE waste under Inventory Module 1. In other words, the number of packages is increased for Inventory Module 1 compared to the Proposed Action, but the content of each individual idealized waste package remains the same.

Table I-5. Abstracted inventory (grams) of radionuclides passing the screening analysis in each idealized waste package for commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste for Proposed Action, Inventory Module 1, and Inventory Module 2.[a,b]

Radionuclide	Commercial spent nuclear fuel waste packages	Codisposal waste packages: DOE spent nuclear fuel	Codisposal waste packages: High-level radioactive waste
Actinium-227	0.00000309	0.000113	0.000467
Americium-241	10,900	117	65.7
Americium-243	1,290	1.49	0.399
Carbon-14	1.37	0.0496	0.00643
Cesium-137	5,340	112	451
Iodine-129	1,800	25.1	48
Neptunium-237	4,740	47.9	72.3
Proactinium-231	0.00987	0.325	0.796
Lead-210	0	0.000000014	0.000000114
Plutonium-238	1,510	6.33	93.3
Plutonium-239	43,800	2,300	3,890
Plutonium-240	20,900	489	381
Plutonium-242	5,410	11.1	7.77
Radium-226	0	0.00000187	0.0000167
Radium-228	0	0.00000698	0.00000319
Strontium-90	2,240	55.4	288
Technetium-99	7,680	115	729
Thorium-229	0	0.0266	0.00408
Thorium-230	0.184	0.0106	0.00782
Thorium-232	0	14,900	7,310
Uranium-232	0.0101	0.147	0.000823
Uranium-233	0.07	214	11.1
Uranium-234	1,830	57.2	47.2
Uranium-235	62,800	8,310	1,700
Uranium-236	39,200	853	39.8
Uranium-238	7,920,000	509,000	261,000

a. Source: DIRS 154841-BSC (2001, Table 36, p. 38).
b. The idealized waste packages in the simulation (model) are based on the inventory abstraction. While the total inventory is represented by the material in the idealized waste packages, the actual number of waste packages emplaced in the potential repository would be different. The numbers of idealized waste packages modeled for various inventory abstractions are listed in Table I-4.

I.3.1.3 Radionuclide Inventory Used in the Model of Long-Term Performance for Inventory Module 2

Wastes with concentrations above Class-C limits (shown in 10 CFR 61.55, Tables 1 and 2) for long and short half-life radionuclides, respectively, are called Greater-Than-Class-C low-level waste. These wastes generally are not suitable for near-surface disposal. The Greater-Than-Class-C waste inventory is discussed in detail in Appendix A, Section A.2.5, of this EIS.

DOE Special-Performance-Assessment-Required low-level radioactive waste could include production reactor operating wastes, production and research reactor decommissioning wastes, non-fuel-bearing components of naval reactors, sealed radioisotope sources that exceed Class-C limits for waste classification, DOE isotope production related wastes, and research reactor fuel assembly hardware. The Special-Performance-Assessment-Required waste inventory is discussed in detail in Appendix A.

The final disposition method for Greater-Than-Class-C and Special-Performance-Assessment-Required low-level radioactive waste is not known. If these wastes were to be placed in a repository, they would be placed in canisters before shipment. This appendix assumes the use of a canister similar to the naval dual-purpose canister described in Section A.2.2.5.6 of Appendix A of this EIS.

IDEALIZED WASTE PACKAGES

The number of waste packages used in the performance assessment simulations do not exactly match the number of waste packages projected for the Proposed Action.

The TSPA model uses two types of *idealized waste packages* (commercial spent nuclear fuel package and co-disposal package), representing the averaged inventory of all the actual waste packages used for a particular waste category.

While the number of idealized waste packages varies from the number of actual waste packages, the total radionuclide inventory represented by all of the idealized waste packages collectively is representative of the total inventory, for the radionuclides analyzed, given in Appendix A of this EIS for the purposes of analysis of long-term performance. The abstracted inventory is designed to be representative for purposes of analysis of long-term performance and cannot necessarily be used for any other analysis, nor can it be directly compared to any other abstracted inventory used for other analyses in this EIS.

Table I-6 lists existing and projected volumes through 2055 for the three Greater-Than-Class-C waste sources. DOE conservatively assumes 2055 because that year would include all Greater-Than-Class-C low-level waste resulting from the decontamination and decommissioning of commercial nuclear reactors. The projected volumes conservatively reflect the highest potential volume and activity expected based on inventories, surveys, and industry production rates.

The data concerning the volumes and projections of Greater-Than-Class-C low-level waste are from Appendix A-1 of the *Greater-Than-Class-C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities, and Other Characteristics* (DIRS 101798-DOE 1994, all). That appendix provides detailed radioactivity reports for such waste currently stored at nuclear utilities.

The only difference between Inventory Modules 1 and 2 is the addition of Greater-Than-Class-C and Special-Performance-Assessment-Required wastes under Inventory Module 2. This represents an incremental increase in the total inventory for Inventory Module 2 over Inventory Module 1, with no difference in the temperature operating mode or the areal extent of the repository disposal area. Because

Table I-6. Greater-Than-Class-C low-level waste volumes (cubic meters)[a] by source.[b]

Source	1993	2055
Nuclear electric utility	26	1,300
Sealed sources	39	240
Other	74	470
Totals	***139***	***2,010***

a. To convert cubic meters to cubic feet, multiply by 35.314.
b. Source: DIRS 101798-DOE (1994, Tables 6-1 and 6-3).

of this, calculations for analysis of long-term performance for Inventory Module 2 were performed considering only Greater-Than-Class-C and Special-Performance-Assessment-Required waste inventories, and the results treated as an incremental increase to the impacts predicted for analysis of long-term performance of Inventory Module 1.

The radionuclide inventory used for Inventory Module 2 (Greater-Than-Class-C and Special-Performance-Assessment-Required materials) is described and tabulated in Appendix A of this EIS and the abstracted per-package inventory developed from these data is listed on Table I-7. The details of

Table I-7. Abstracted inventory (grams) of radionuclides passing the screening analysis in each idealized waste package for Greater-Than-Class-C and Special-Performance-Assessment-Required waste (grams per waste package) under Inventory Module 2.[a,b]

Radionuclide	Greater-Than-Class-C and Special-Performance-Assessment-Required	Radionuclide	Greater-Than-Class-C and Special-Performance-Assessment-Required
Actinium-227	0	Plutonium-242	0.00614
Americium-241	40	Radium-226	0.0504
Americium-243	0.00151	Strontium-90	0.82
Carbon-14	28.9	Technetium-99	568
Cesium-137	771	Thorium-229	0
Iodine-129	0.000705	Thorium-230	0
Neptunium-237	0	Uranium-232	0.00000287
Protactinium-231	0	Uranium-233	0.00419
Lead-210	0	Uranium-234	0
Plutonium-238	1.56	Uranium-235	0
Plutonium-239	2,860	Uranium-236	0
Plutonium-240	0.0123	Uranium-238	563,000
Plutonium-241	0.0207		

a. Source: DIRS 157307-BSC (2001, Enclosure 1).
b. The idealized waste packages in the simulation (model) are based on the inventory abstraction. While the total inventory is represented by the material in the idealized waste packages, the actual number of waste packages emplaced in the potential repository would be different.

obtaining the per-package inventory for Greater-Than-Class-C and Special-Performance-Assessment-Required materials are described in DIRS 157307-BSC (2001, Attachment III).

I.3.2 INVENTORY FOR WATERBORNE CHEMICALLY TOXIC MATERIALS

Waterborne chemically toxic materials that present a potential human health risk would be present in materials disposed of in the repository. The most abundant of these materials would be nickel, chromium, and molybdenum (which would be used in the waste package) and uranium (in the disposed waste). Uranium is both a chemically toxic and a radiological material. Screening studies were conducted to determine if any of these or other materials would be released in sufficient quantities to have a meaningful impact on groundwater quality.

An inventory of chemical materials to be placed in the repository under the Proposed Action was prepared. The inventories of the chemical components in the repository were combined into four groups:

- Materials not part of engineered barrier system
- Components of the engineered barrier system including:
  - Titanium drip shields
  - Alloy-22 in the outer layer of the waste packages
  - Stainless steel in the inner layer of the waste packages
- Other materials internal to the waste packages
- High-level radioactive waste

These materials were organized into groups with similar release times for use in the screening study. Table I-8 lists the chemical inventories. Plutonium is not listed in Table I-8 because, while it is a heavy metal and therefore could have toxic effects, its radiological toxicity far exceeds its chemical toxicity (DIRS 102205-DOE 1998, Section 2.6.1). In addition, while there are radiological limits set for exposure

Table I-8. Inventory (kilograms)[a] of chemical materials placed in the Proposed Action repository.

Element	Inventory: Not part of engineered barrier system	Engineered barrier system components exposed before waste package failure	Internal to waste package including inner sleeve	High-level radioactive waste[b]	Totals
Aluminum	0	0	2,452,400	0	2,452,400
Barium	0	0	50,000	74,000[c]	124,000
Boron	0	0	197,400	0	197,400
Cadmium	0	0	3,400	43,000	46,400
Carbon	318,738	547	5,000	0	324,285
Chromium	0	23,735,000	26,414,000	0	50,149,000
Cobalt	0	0	27,000	0	27,000
Copper	243,800	0	3,000	0	246,800
Iron	111,916,880	1,190,000	161,695,000	0	274,801,880
Lead	0	0	0	2,000	2,000
Magnesium	0	0	12,000	0	12,000
Manganese	1,189,576	575,880	3,732,100	0	5,497,556
Mercury	0	0	0	200	200
Molybdenum	0	17,307,000	3,839,100	0	21,146,100
Nickel	0	60,797,000	18,659,100	0	79,456,100
Phosphorus	39,842	820	91,200	0	131,862
Selenium	0	0	0	300	300
Silicon	330,122	18,226	1,680,500	0	2,028,848
Sulfur	39,842	547	68,200	0	108,589
Titanium	0	4,148,000	2,000	0	4,150,000
Uranium	0	0	70,000,000	0	70,000,000
Vanadium	0	377,600	0	0	377,600
Zinc	0	0	3,000	0	3,000

a. To convert kilograms to pounds, multiply by 2.2046

b. The high-level radioactive waste form to be placed in the potential repository would not exhibit the Characteristic of Toxicity as measured by the Toxicity Characteristic Leaching Procedure (40 CFR 261.24).

c. Includes barium grown in from decay of all of the cesium.

to plutonium, no chemical toxicity benchmarks have been developed for this element. Therefore, lacking data to analyze chemical toxicity, plutonium was not analyzed for the chemical screening.

I.3.3 INVENTORY FOR ATMOSPHERIC RADIOACTIVE MATERIALS

The only radionuclide that would have a relatively large inventory and a potential for gas transport would be carbon-14. Iodine-129 can exist in a gas phase, but it is highly soluble and therefore likely to dissolve in groundwater rather than migrate as a gas. Radon-222 is a gas, but would decay to a solid isotope before escaping from the repository region (see Section I.7.3). After carbon-14 escaped from the waste package, it could flow through the rock in the form of carbon dioxide. About 2 percent of the carbon-14 in commercial spent nuclear fuel occurs in a gas phase in the space (or *gap*) between the fuel and the cladding around the fuel (DIRS 103446-Oversby 1987, p. 92). The gas-phase inventory consists of 0.122 curie of carbon-14 per commercial spent nuclear fuel waste package. Table I-9 lists the carbon-14 inventory for commercial spent nuclear fuel under the Proposed Action and Inventory Modules 1 and 2.

I.4 Extension of TSPA Methods and Models for EIS Analysis of Long-Term Performance

The TSPA model nominal scenario is equivalent to the Proposed Action inventory for an individual at the RMEI location [approximately 18 kilometers (11 miles) downgradient from the proposed repository]. Details on the adaptations, extensions, and modifications to the software and models used for the TSPA

Table I-9. Carbon-14 gaseous inventory from commercial spent nuclear fuel (curies).[a]

Modeled inventory	Quantity[b]
Proposed Action	959
Module 1	1,434
Module 2	1,434

a. Impacts of carbon-14 in solid form are addressed as waterborne radioactive material impacts.

b. Based on 0.122 curie of carbon-14 per commercial spent nuclear fuel waste package, based on 2 percent of the carbon-14 in commercial spent nuclear fuel existing as a gas in the gap between the fuel and the cladding around the fuel (DIRS 103446-Oversby 1987, p. 92).

model necessary to analyze impacts under Inventory Modules 1 and 2 and at additional individual locations 30 and 60 kilometers (19 and 37 miles) downgradient from the repository are presented in this section.

I.4.1 METHODOLOGY

The calculations presented in this EIS were performed using the numerical code GoldSim, Version 7.17.200 (DIRS 155182-BSC 2001, all). The GoldSim calculations were performed for the conceptual/ process modeling of the proposed Yucca Mountain Repository described in the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, all) and expanded upon in the *FY01 Supplemental Science and Performance Analyses* (DIRS 155950-BSC 2001, all; DIRS 154659-BSC 2001, all).

The performance assessment calculations for both the TSPA–Site Recommendation, the Supplemental Science and Performance Analyses, and the analysis of long-term performance calculations described in this EIS were performed within a probabilistic framework combining the most likely ranges of behavior for the various component models, processes, and corresponding parameters included in the overall conceptual/process model describing the performance of the repository.

The GoldSim software integrated the submodels using a Monte-Carlo simulation-based methodology to create multiple random combinations of the uncertain variables, and computed the probabilistic performance of the entire waste-disposal system in terms of annual individual dose. The GoldSim software calculated radionuclide release and radiological dose (the annual committed effective dose equivalent as defined in 40 CFR 197.2 from individual radionuclides and the total annual dose due to all radionuclides released from the repository from failed waste packages). In this EIS, the annual committed effective dose equivalent is referred to as the annual individual dose. GoldSim calculated the total annual dose for 300 realizations of the model configuration for the nominal scenario, and 5,000 realizations for the igneous activity scenario, using randomly selected values of distributed parameters for each realization. The calculation results are available in two main forms: (1) probability distributions for peak dose to an individual, and (2) time histories of annual dose to an individual.

The recently promulgated Environmental Protection Agency Final Rule 40 CFR Part 197 stipulates that the performance assessment of the proposed Yucca Mountain Repository include an estimate of dose to the reasonably maximally exposed individual. The Rule further states that this assessment provide, for 10,000 years, the reasonably maximally exposed individual annual committed effective dose equivalent (40 CFR 197.20 and 197.25). For the purposes of this EIS, the analysis of long-term performance must calculate the peak dose that would occur within the period of geologic stability (40 CFR 197.35). The peak dose is projected to occur within 1,000,000 years.

The methodology used for the calculations presented in this EIS draws upon the extensive analyses carried out in support of the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, all) and in the *FY01 Supplemental Science and Performance Analyses,* Volumes 1 and 2 (DIRS 155950-BSC 2001, all, and DIRS 154659-BSC 2001, all). Only those model components and related parameters that were modified to account for the scenarios considered in addition to those used in the TSPA–Site Recommendation or the *FY01 Supplemental Science and Performance Analyses* are described in this EIS. In addition, the model configuration used for the calculations presented in this EIS was modified to conform to the recently promulgated U.S. Environmental Protection Agency Final Rule. The Final Rule provides the criteria to be used in determining the RMEI location [40 CFR 197.21(a)], the other criteria of the RMEI (that were applied in the calculation of biosphere dose-conversion factors), and the groundwater protection standard, including the representative volume to be used for the calculation of gross alpha activity, total radium activity, and whole-body dose (40 CFR 197.30, Table 1). These modifications are described in Section I.4.4.

This EIS considers inventories in addition to those described in the TSPA–Site Recommendation and the *Supplemental Science Performance Analyses* for the 70,000 MTHM inventory. The calculations in this EIS include the Proposed Action (70,000 MTHM inventory) under both the higher-temperature repository operating mode and lower-temperature operating mode, and the Module 1 and 2 inventories under the higher-temperature operating mode, for the following scenarios:

- The nominal scenario that considers performance of the repository under undisturbed conditions, but including seismic activity.

- The human intrusion scenario (DIRS 153246-CRWMS M&O 2000, Section 4.4, pp. 4-25 to 4-32), that considers an "intruder" drilling a land-surface borehole using a drilling apparatus (under the common techniques and practices currently employed in exploratory drilling for groundwater in the region around Yucca Mountain), drilling directly through an intact or degraded waste package, and subsequently into the uppermost aquifer underlying Yucca Mountain. The intrusion then causes the subsequent compromise and release of contaminated material in the waste package. The human-intrusion scenario was simulated for a 1-million year performance period with the intrusion at 30,000 years after repository closure.

- The igneous activity scenario contains two separate possible events: a volcanic eruption that includes exposure as a result of atmospheric transport and deposition on the ground, and an igneous intrusion groundwater transport event (DIRS 155950-BSC 2001, Section 14.2.1, p. 14-5). In the volcanic eruption event (DIRS 153246-CRWMS M&O 2000, Section 3.10, pp. 3-187 to 3-216), a dike (or dikes) would intersect the repository and compromise all waste packages in the conduit. Then, an eruptive conduit of an associated volcano would intersect waste packages in its path. Waste packages in the path of the conduit would be sufficiently damaged that they provide no further protection, and the waste in the packages would be entrained in the eruption and subject to atmospheric transport. In the igneous intrusion groundwater transport event, the analysis calculated releases caused by a dike (or dikes) intersecting emplacement drifts, causing varying degrees of waste-package damage.

I.4.2 ASSUMPTIONS

This section identifies assumptions that are essential for this calculation. The assumptions listed here contribute to the generation of results reported in Sections I.5 and I.6 of this appendix.

1. The Proposed Action (70,000-MTHM) model configuration for the calculations in this EIS consists of the *FY01 Supplemental Science and Performance Analyses* model (DIRS 155950-BSC 2001, all), which differs from the TSPA–Site Recommendation model (DIRS 153246-CRWMS M&O 2000, all).

The model used for the calculations in Sections I.5 below includes the modifications from the Supplemental Science and Performance Analyses and TSPA–Site Recommendation models as described below in Section I.4.4. Other assumptions incorporated into the Supplemental Science and Performance Analyses model are documented in the *FY01 Supplemental Science and Performance Analyses* Volume 2 (DIRS 154659-BSC 2001, Section 2, all). The key differences between the Supplemental Science and Performance Analyses and the model configuration used in the calculations presented in this EIS are described in Section I.4.4.

2. The radionuclide inventories used in the calculations in Section I.5 are those developed in the *Inventory Abstraction* Analysis Model Report (DIRS 154841-BSC 2001, Table 36, p. 38). The per-waste-package inventories for commercial spent nuclear fuel and codisposal waste packages are the same as those used in the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, Section 3.5.1, pp. 3-94 to 3-100), but the DOE-owned spent nuclear fuel inventory does not include naval spent nuclear fuel. The naval spent nuclear fuel is conservatively represented by commercial spent nuclear fuel (DIRS 152059-BSC 2001, all, and DIRS 153849-DOE 2001, Section 4.2.6.3.9, p. 4-257). The per-waste-package inventories used for the Greater-Than-Class-C calculations use the Greater-Than-Class-C inventory presented in Attachment VI of the *EIS Performance–Assessment Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain* (DIRS 155393-CRWMS M&O 2000, p. VI-5) divided according to the number of packages indicated in DIRS 155393-CRWMS M&O (2000, Attachment VI).

I.4.3 USE OF COMPUTER SOFTWARE AND MODELS

The calculations described in this EIS were performed using the numerical code GoldSim, Version 7.17.200 (DIRS 155182-BSC 2001, all). GoldSim was developed by Golder Associates as an update to the baseline software RIP v.5.19.01 (DIRS 151395-Golder Associates 1998, all). GoldSim is an object-oriented program that is computationally similar to RIP v.5.19.01, which was used for the TSPA–Viability Assessment (DIRS 101779-DOE 1998, Volume 3, p. 2-29). GoldSim is designed such that probabilistic simulations can be conducted and represented graphically.

I.4.4 MODIFICATIONS TO THE TSPA–SITE RECOMMENDATION AND SUPPLEMENTAL SCIENCE AND PERFORMANCE ANALYSIS MODELS

This EIS builds on the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, all) and *FY01 Supplemental Science and Performance Analyses* (DIRS 155950-BSC 2001, all) modeling of the Proposed Action (70,000-MTHM) repository configuration. Because this EIS evaluates the possible consequences of ultimately including the entire commercial spent nuclear fuel, DOE-owned spent nuclear fuel, and high-level radioactive waste inventories, an expanded repository area was also considered.

The change from the TSPA–Site Recommendation waste inventory and repository area to a calculation of the performance of an expanded repository includes addition of the Lower Block, shown on Figure I-3, in the calculations. The TSPA–Site Recommendation and Supplemental Science and Performance Analyses reports relied only on a detailed analysis of just the Primary Block shown on Figure I-3.

The GoldSim numerical code simulates transport of radionuclides from the repository, through the unsaturated zone, and through the saturated zone to the accessible environment. The different process models included in the GoldSim code are fully described and documented in the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, all). The unsaturated zone transport release nodes and saturated zone transport capture areas for the 70,000-MTHM inventory in the TSPA–Site Recommendation and Supplemental Science and Performance Assessment models were modified for Inventory Modules 1 and 2 to include the Lower Block emplacement area (DIRS 157307-BSC 2001, all).

The GoldSim model configuration used for the Supplemental Science and Performance Analyses was modified to conform to the recently published Environmental Protection Agency Final Rule 40 CFR Part 197. The model also assesses the performance of additional radionuclide inventories and performance scenarios. Sections I.4.4.1 through I.4.4.8 describe the modifications to the TSPA–Site Recommendation and Supplemental Science and Performance Analyses models. The model configuration for the calculations in this EIS differs from earlier performance assessment models in the following areas:

- The model used for the calculations in this EIS used biosphere dose conversion factor based on the RMEI defined in 40 CFR 197.21. The models used in the TSPA–Site Recommendation and Supplemental Science and Performance Analyses used different biosphere dose conversion factors based on the average member of the critical group in the then-proposed 10 CFR 63.115.

- The length of the saturated zone simulated in the model configuration for the calculations in this EIS extends from inside the repository footprint to latitude 36 degrees 40 minutes 13.6661 seconds north, above the highest concentration of radionuclides in the plume of contamination. The RMEI is assumed to reside at this location in the accessible environment. The latitude at this location is at the southwestern corner of the Nevada Test Site.

- Groundwater protection was assessed using an annual representative volume of 3.7 million cubic feet (exactly 3,000 acre-feet) per year of groundwater, as specified at 40 CFR Part 197, to calculate the total alpha activity, the total radium concentration, and the whole-body dose. To calculate all other concentrations not included in the groundwater-protection standard, the water usage was assigned in the same probabilistic manner used in the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, Section 3.9.2.4, p. 3-184) and the *FY01 Supplemental Science and Performance Analyses* (DIRS 155950-BSC 2001, Section 13.3.5, pp. 13-41 to 13-44).

- The waste inventories used for the calculations in this EIS are presented in DIRS 154841-BSC (2001, Table 36, p. 38).

- Waste-package corrosion for the calculations in this EIS would be due to general corrosion independent of temperature, as was true for the TSPA–Site Recommendation. The Supplemental Science and Performance Analyses calculations included temperature-dependent waste package corrosion.

- The process-level lower-temperature operating mode thermal-hydrologic results for this EIS were corrected from those presented in the Supplemental Science and Performance Analyses to include radiative heat transfer in the unsaturated zone modeling with the Nonisothermal Unsaturated-Saturated Flow and Transport model.

I.4.4.1 Modifications to FEHM Particle Tracker Input and Output

The unsaturated zone flow-and-transport modeling in the TSPA–Site Recommendation, in the *FY01 Supplemental Science and Performance Analyses*, and in this EIS are conducted with the Finite Element Heat and Mass (FEHM) model. The movement of fluid and radionuclides released from the waste packages was modeled in the unsaturated zone by means of a particle-tracking algorithm in the TSPA–Site Recommendation and Supplemental Science and Performance Analyses process models (DIRS 153246-CRWMS M&O 2000, p. 2-27; DIRS 155950-BSC 2001, Section 11). The particle-tracking files used in the TSPA–Site Recommendation were modified for the increased inventories of Modules 1 and 2 to allow the FEHM unsaturated zone input regions to correspond to the Lower Block area used for the simulations. The interface file in GoldSim was modified for this case by changing the FEHM nodes used for transport from the Primary Block as considered in the TSPA–Site Recommendation and the Supplemental Science and Performance Analyses for the Proposed Action inventory. The calculations presented in this EIS also include the Lower Block of a potential repository that would also be used for input of mass from an expanded repository area. The FEHM nodes were chosen to correspond to the Lower Block repository coordinates because of the changes to the regions from where mass is captured coming out of the FEHM model (DIRS 155393-CRWMS M&O 2000, Attachments II and III). Capture regions at the surface of the saturated zone would accumulate water and mass released from the repository that had been transported through the unsaturated zone. The capture regions for the Primary Block are shown in Figure I-4. These capture regions were modified to ensure all the mass would be captured and to distribute the mass to the saturated zone capture regions, including release from the Lower Block. Figure I-5 shows the capture regions used for Inventory Modules 1 and 2.

The repository nodes were extracted based on the information and representation of the repository configuration described in *EIS Performance-Assessment Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain* (DIRS 155393-CRWMS M&O 2000, Attachment II). The drifts in the Lower Block were first aggregated into larger groups based on similar elevations. Then the boundary coordinates of the larger groups were used to define rectangular regions. The Software Routine *repocoord1.f* (Version 1.0) (DIRS 155393-CRWMS M&O 2000, Attachment II) was used to extract FEHM nodes within the rectangular region. The extracted nodes were then plotted using SigmaPlot (Version 4.01, a commercial graphics software package) and nodes that fell beyond the defined drift region were removed from the repository node list. The use of *repocoord1.f* (Version 1.0) in these EIS calculations is documented in DIRS 155393-CRWMS M&O (2000, Attachment II).

I.4.4.2 Estimation of the Thermal Profiles and Infiltration for the Lower Block

The TSPA–Site Recommendation and *FY01 Supplemental Science and Performance Analyses* models used to assess repository performance utilized thermal-hydrologic modeling to estimate infiltration from land surface to the repository horizon. Infiltration water would be the principal cause of waste-package corrosion and the main agent for waste transport. The TSPA–Site Recommendation and *FY01 Supplemental Science and Performance Analyses* model conceptualizations for the Yucca Mountain Repository would be considered waste forms in discrete areal regions of the repository as source terms for flow and transport from the repository to the saturated zone. The GoldSim conceptualization for the TSPA–Site Recommendation considered the repository block, referred to as the Primary Block, to be comprised of four source regions (Figure I-4). The four regions are covered by the Yucca Mountain multiscale thermohydrologic model and its abstraction, which was used to develop the thermodynamic-environment time histories at different potential waste-package locations distributed throughout the Primary Block (DIRS 139610-CRWMS M&O 2000, Section 6.6, all; DIRS 154594-CRWMS M&O 2001, Section 6.3). These time histories for the higher-temperature operating mode were used in both the TSPA–Site Recommendation and the *FY01 Supplemental Science and Performance Analyses*.

The calculations for Inventory Modules 1 and 2 for this EIS used two additional areas for disposal, using an additional approximately 0.88 square kilometer (218 acres) of the Primary Block that was not used in the design of the Primary Block for the Proposed Action, the higher-temperature operating mode (DIRS 150941-CRWMS M&O 2000, Figure 4-14), and approximately 1.7 square kilometers (408 acres) of the Lower Block, which would be to the east of the Primary Block (Figure I-3). For Inventory Modules 1 and 2, source region 2 was expanded to the east so that its areal extent would include the Lower Block (Figure I-5) (DIRS 155393-CRWMS M&O 2000, Section 5.2.2, p. 11-12).

The following methodology was used to develop thermal histories for waste packages emplaced in the Lower Block. The thermal response from the multiscale thermohydrologic model (DIRS

149862-CRWMS M&O 2000, all) is correlated to the distance from the edge of the repository. Further, seepage into the drift would be a function of the local infiltration flux. Therefore, the location and estimated infiltration flux were used to select analogous Primary Block thermal-hydrologic responses for application to comparable locations in the Lower Block. Thus, the Primary Block thermal-hydrologic data were extended to the 51 Lower Block elements shown on Figure I-6. It should be noted that DOE would pursue a comprehensive characterization of these blocks before it used them for waste emplacement. The modeling work described in this EIS related to these uncharacterized blocks is limited to estimating the environmental impacts under the expanded inventory (Modules 1 and 2) configuration. The detail on extending this method to the 51 nodes is in DIRS 155393-CRWMS M&O (2000, Attachment II, pp. II-2 to II-5), and the estimation of lower-block infiltration seepage rates is in DIRS 155393-CRWMS M&O (2000, Attachment III, pp. III-2 to III-19). The glacial-transition climate infiltration rate for the 51 elements was estimated from the site-scale hydrologic model (DIRS 100103-Bodvarsson, Bandurraga, and Wu 1997, all). For each of the 51 Lower Block elements, the GoldSim code was configured with thermal history data sets from the site multiscale thermohydrologic model (DIRS 139610-CRWMS M&O 2000, Section 6.6, all, and its abstractions; DIRS 154594-CRWMS M&O 2001, Section 6.3) based on similar infiltration and proximity to the edge of the repository as the analogous Primary Block locations. Using these data, the infiltration categories, or bins, for the waste packages associated with the Inventory Modules 1 and 2 cases were established as described in Attachment IV of the calculation document *EIS Performance-Assessment Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain* (DIRS 155393-CRWMS M&O 2000, pp. IV-2 to IV-4). The use of thermal profiles in estimating infiltration to the repository blocks is described in detail in Attachment III of the calculation document *EIS Performance-Assessment Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain* (DIRS 155393-CRWMS M&O 2000, pp. III-2 to III-19). DIRS 157307-BSC (2001, Attachment II) describes the calculation of the fractional Lower Block repository areas corresponding to the infiltration bins for these calculations.

I.4.4.3 Saturated Zone Breakthrough Curves

Transport in the saturated zone beneath the repository would be the main route for groundwater transport of contaminants leached from the repository. The radioactive contaminants would move through the saturated zone to the accessible environment. The accessible environment is defined as any area outside the controlled area (40 CFR 197.12). The Environmental Protection Agency Final Rule (40 CFR 197.12) specifies the following elements of the controlled area:

1. The surface area, identified by passive institutional controls, that encompasses no more than 300 square kilometers (about 74 acres). It must not extend farther:
   a. South than 36 degrees 40 minutes 13.6661 seconds north latitude, in the predominant direction of groundwater flow; and
   b. Than 5 kilometers (3 miles) from the repository footprint in any other direction; and
2. The subsurface underlying the surface area.

The location where the RMEI would reside, where groundwater protection was analyzed, would be the point above the highest concentration of radionuclides in the simulated plume of saturated zone contamination where the plume crossed the southernmost boundary of the controlled area (at a latitude of 36 degrees 40 minutes 13.6661 seconds North) and reached the accessible environment. For this analysis, DOE selected the southern boundary of the controlled area and the location of the RMEI to be at the limit discussed above, which is approximately 18 kilometers (11 miles) from the potential repository, compared to the corresponding distance of approximately 20 kilometers (12 miles) used in the saturated zone transport modeling for TSPA–Site Recommendation and the *FY01 Supplemental Science and*

Performance Analyses, as shown in Figure I-7. To analyze long-term performance with respect to the standard set in the Environmental Protection Agency Final Rule 40 CFR 197.12, additional saturated zone breakthrough curves, which describe the time-related arrivals of radionuclides at the RMEI location, were calculated for all radionuclides used in the calculations in this EIS. The saturated zone breakthrough curves were used in the analyses to simulate radionuclide transport from the water table beneath the proposed repository to the receptor location. Depending on the subsurface layout of a repository, the distance to the RMEI location from any point in the subsurface layout could be more or less than 18 kilometers. For convenience and consistency with other documents, the RMEI location is consistently discussed as being approximately 18 kilometers (11 miles) downgradient from the proposed repository.

To generate the saturated zone breakthrough curves used in the calculations in this EIS, 100 realizations of the saturated zone site-scale flow-and-transport model were performed as described for the saturated zone process model (DIRS 139440-CRWMS M&O 2000, Sections 6.2 and 6.3) to generate saturated zone breakthrough curves at the RMEI location. Other stochastic parameters for the saturated zone simulations use the same values as those used in the saturated zone breakthrough curves for the *FY01 Supplemental Science and Performance Analyses* (DIRS 154659-BSC 2001, Section 3.2.10). The simulated radionuclide breakthrough curves at the RMEI location exhibited shorter transport times than those at 20 kilometers (12 miles), as presented in *Supplemental Sciences and Performance Analyses* (DIRS 155950-BSC 2001, Section 13.2.1.3) on a realization-by-realization basis. In particular, radionuclides that could have significantly greater sorption in the alluvium than in the volcanic units (such as neptunium-237) exhibited shorter transport times to the RMEI location in this analysis relative to the 20-kilometer location used in the TSPA–Site Recommendation, the *Supplemental Science Performance Analyses*, and in the Draft EIS. This result is related to the fact that the RMEI location in this analysis would result in a decrease in the length of transport through the alluvium relative to the transport path to the 20-kilometer location.

The approach used for simulations of groundwater flow and radionuclide transport in the saturated zone used in this EIS is the same as the approach used in the TSPA–Site Recommendation. The saturated zone site-scale flow-and-transport model was used to simulate the unit radionuclide mass breakthrough curves for radionuclides of concern to the Site Recommendation at the RMEI location. In the model configuration for the calculations for this EIS, these saturated zone breakthrough curves are coupled with the other components of the system (mass flux and representative volume or water usage) using the convolution-integral method in the same manner as described and implemented in the GoldSim program for the TSPA–Site Recommendation and the *FY01 Supplemental Science and Performance Analyses* (DIRS 153246-CRWMS M&O 2000, Section 2.2.2; DIRS 155950-BSC 2001, Section 3.2.10). In addition, the simulation of radionuclide decay chains and the transport of decay products in the saturated zone system was performed using a one-dimensional model directly in the GoldSim numerical code.

In the saturated zone model, the capture regions that would accumulate flow and mass at the base of the unsaturated zone become the source regions for the saturated zone model. The four radionuclide source regions in the saturated zone (Figures I-4 and I-5) that were defined for the 70,000-MTHM case of the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, Section 3.8.2.2 and Figure 3.8-14, p. F3-117) were used in the calculations in this EIS. For Inventory Modules 1 and 2, radionuclide mass originating from the Lower Block of the repository is applied to source region number 2 in the saturated zone transport module. The Lower Block of the expanded repository would extend farther to the east than the saturated zone source region number 2 for the TSPA–Site Recommendation base case. However, applying the radionuclide mass from the Lower Block to this source region constitutes a conservative approximation of transport in the saturated zone. Lower permeability rocks of the upper volcanic confining unit exist at the water table in the area immediately to the east of saturated zone source region number 2, which would result in slower initial advective groundwater velocity for radionuclide transport in this area. Preliminary results of radionuclide transport simulations with the saturated zone site-scale flow and transport model confirm that radionuclide transport times in the saturated zone from the area below the Lower Block would be longer than the transport times from saturated zone source region number 2 in the Proposed Action considered in this EIS.

I.4.4.4 Modification to the Waste Package Degradation Model

The WAste Package DEGradation (WAPDEG) model (DIRS 151566-CRWMS M&O 2000, all) was used to calculate drip shield and waste package degradation profiles with time in the GoldSim TSPA model configurations used for TSPA–Site Recommendation, *FY01 Supplemental Science and Performance Analyses*, and this EIS. Several input parameters to the WAPDEG model developed for TSPA–Site Recommendation (DIRS 151566-CRWMS M&O 2000, all) were reevaluated in the *FY01 Supplemental Science and Performance Analyses*, Volume 1 (DIRS 155950-BSC 2001, Section 7). The reevaluation led to the following changes to the TSPA–Site Recommendation WAPDEG model and parameters used in the *FY01 Supplemental Science and Performance Analyses* and the calculations in this EIS. These changes are described in detail in *FY01 Supplemental Science and Performance Analyses*, Volume 1 (DIRS 155950-BSC 2001, Section 7) and are summarized here:

- All surface-breaking weld flaws and all weld flaws embedded in the outer one quarter of the closure weld thickness were considered capable of propagation in the radial direction in the WAPDEG model developed for the TSPA–Site Recommendation (DIRS 151566-CRWMS M&O 2000, Section 5.5, p. 39). In the *FY01 Supplemental Science and Performance Analyses* and in this analysis, the fraction of these weld flaws capable of propagation in the radial direction is given by a ±3 standard deviation truncated lognormal distribution with a mean of 0.01 and bounded between 0.5 (+3 standard deviations) and 0.0002 (-3 standard deviations) (DIRS 155950-BSC 2001, Section 7.3.3.3.4, p. 7-41).

- The stress threshold for the initiation of stress corrosion cracking was given by a uniform distribution between 20 and 30 percent of the Alloy-22 yield strength in the WAPDEG model developed for the TSPA–Site Recommendation (DIRS 151566-CRWMS M&O 2000, Section 4.1.9, p. 29). In the *FY01 Supplemental Science and Performance Analyses* and for this analysis, the stress threshold for the initiation of stress corrosion cracking is given by a uniform distribution between 80 and 90 percent of the Alloy 22 yield strength (DIRS 155950-BSC 2001, Section 7.3.3.3.3, p. 7-39).

- The uncertainty bounds of the residual stress profile in the Alloy-22 waste package outer closure lid weld regions (induction annealed) were set to ±30 percent of the yield strength of Alloy-22 in the WAPDEG Model developed for TSPA–Site Recommendation (DIRS 151566-CRWMS M&O 2000, Section 6.5.1, p. 79). In the *FY01 Supplemental Science and Performance Analyses* and in this analysis, the uncertainty bounds of the residual stress profile in the Alloy-22 waste package outer closure lid weld regions were set to ±21.4 percent of the yield strength (DIRS 155950-BSC 2001, Section 7.3.3.3.1, p. 7-74).

- The uncertainty bounds of the residual stress profile in the Alloy-22 waste package inner closure lid weld regions (laser peened) were set to ±30 percent of the yield strength of Alloy-22 in the WAPDEG model developed for TSPA–Site Recommendation (DIRS 151566-CRWMS M&O 2000, Section 6.5.1, p. 79). In the *FY01 Supplemental Science and Performance Analyses* and in this analysis, the uncertainty bounds of the residual stress profile in the Alloy-22 waste package inner closure lid weld regions were sampled from a cumulative distribution function (DIRS 155950-BSC 2001, Section 7.3.3.3.2, p. 7-37 and Table 7.3.3-2, p. 7T-4).

- The variances of the general corrosion rate distributions for Alloy-22 and titanium Grade 7 were considered to result from contributions of both uncertainty and variability in the WAPDEG model developed for the TSPA–Site Recommendation (DIRS 151566-CRWMS M&O 2000, Section 6.3.1, p. 55). In *FY01 Supplemental Science and Performance Analyses* and in this analysis, the total variance of the general corrosion rate distributions was treated as uncertainty (DIRS 155950-BSC

2001, Section 7.3.5.2, p. 7-54). To ensure conservatism in the analysis, the temperature-dependent Alloy-22 general corrosion model developed for the *FY01 Supplemental Science and Performance Analyses* (DIRS 155950-BSC 2001, Section 7.3.5.3.2, p. 7-56) was not used in this analysis. This is conservative because the non-temperature-dependent model uses a high bounding rate characteristic of high temperature, while the temperature-dependent model would take credit for long periods of lower temperatures and corresponding low corrosion rates. The same Alloy-22 and titanium Grade 7 general corrosion rate distributions used in the WAPDEG model developed for TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000. Section 3.4.1, pp. 3-80 to 3-87) and the *FY01 Supplemental Science and Performance Analyses* (DIRS 155950-BSC 2001, Section 7.3.5, pp. 7-52 to 7-61) were also used in the calculations in this EIS. The calculated means of the general corrosion rate distribution used for the calculations in this EIS are 1.94×10^{-4} millimeter (7.64×10^{-6} inch) per year for titanium Grade 7 and 6.80×10^{-5} millimeter (2.68×10^{-6} inch) per year for Alloy-22. The data used to calculate the means are from complementary distribution functions in *WAPDEG Analysis of Waste Package and Drip Shield Degradation* (DIRS 151566-CRWMS M&O 2000, Section 4, pp. 19 and 20).

I.4.4.5 Early Waste Package Failure

The potential waste package early failure mechanisms were reevaluated in the *FY01 Supplemental Science and Performance Analyses*, particularly improper heat treatment of waste packages (DIRS 155950-BSC 2001, Section 7.3.6, p. 7-62). These results are incorporated in the calculations in this EIS. The probability of having one or more waste packages in the repository improperly heat-treated is provided in Table I-3.

In evaluating the potential consequences of early failures by improper heat treatment for the *FY01 Supplemental Science and Performance Analyses* and this EIS, early waste-package failure would occur on initiation of corrosive processes and would be due to failure of the outer and inner closure lids of the waste package outer barrier and the failure of the closure lid of the stainless-steel structural waste package inner shell. Details of the use of this model in performance assessment analyses are discussed in *FY01 Supplemental Science and Performance Analyses*, Volume 2 (DIRS 154659-BSC 2001, Section 3.2.5.4, p. 3-21). The following elements were employed in that evaluation:

1. Those waste packages affected by early waste-package failure would fail immediately by general corrosion as patches (DIRS 154659-BSC 2001, Section 3.2.5.4, p. 3-21).

2. The area on the waste package affected by improper heat treatment would be equal to the area of closure-weld patches because improper heat treatment would be most likely to occur during the induction annealing of the outer closure lid welds of the waste-package outer barrier.

3. The materials of the entire affected area would be lost on failure of the waste packages because the affected area would be subject to stress-corrosion cracking and highly enhanced localized and general corrosion.

4. The weld region of the inner closure lid of the outer barrier and the closure lid of the stainless-steel structural inner shell would fail at the same time the outer closure-lid weld region failed.

These assumptions are conservative because only the weld region of the outer lid of the outer barrier would be affected by potential improper heat treatment during the stress mitigation heat treatment (induction annealing), and the inner lid of the outer barrier would be unlikely to be affected. In a more realistic scenario, the breached weld patches of the affected waste package would remain with the waste package until the weakened areas were affected by a major mechanical impact or corroded away by general corrosion.

I.4.4.6 Biosphere Dose Conversion Factors for the 40 CFR 197 Reasonably Maximally Exposed Individual

Biosphere dose conversion factors were used to estimate the radiation dose that would be incurred by an individual when a unit activity concentration of a radionuclide reached the accessible environment. The biosphere dose conversion factors for the RMEI were developed using the environmental and agricultural parameters characteristic of the Amargosa Valley region, and the dietary and lifestyle characteristics of the RMEI consistent with those specified in 40 CFR 197.21. The lifestyle characteristics of the RMEI were representative of a rural-residential population. The dietary characteristics of the RMEI were based on a food consumption survey (DIRS 100332-DOE 1997, all) for the population of the town of Amargosa Valley, Nevada. Consistent with the final rule at 40 CFR 197.21, the dietary characteristics of the RMEI were represented by the mean values of locally produced food for Amargosa Valley residents. The dietary and lifestyle attributes of the RMEI are listed in Table I-10. The dietary attributes were developed using the set of recently reevaluated and updated values of consumption rates of locally produced food in *Calculation: Consumption Rates of Locally Produced Food in Nye and Lincoln Counties* (DIRS 156016-BSC 2001, all). This set of consumption rates is different from the set used in the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, Section 3.9) and the *FY01 Supplemental Science and Performance Analyses* (DIRS 155950-BSC 2001, Section 13) analyses. The changes include the update of the contingent average daily intakes of food, adjustments in the grouping of the food categories, and adjustments in the selection of individuals whose consumption rates were used to develop the RMEI.

Table I-10. Average values of the dietary and lifestyle attributes for the RMEI.

Parameter	Mean value of the attribute
Leafy vegetables consumption rate (kilograms[a] per year)	3.9
Other vegetables consumption rate (kilograms per year)	4.8
Fruit consumption rate (kilograms per year)	12.4
Grain consumption rate (kilograms per year)	0.3
Meat consumption rate (kilograms per year)	2.6
Poultry consumption rate (kilograms per year)	0.4
Milk consumption rate (liters[b] per year)	4.8
Eggs consumption rate (kilograms per year)	5.6
Fish consumption rate (kilograms per year)	0.3
Water consumption rate (liters per year)	730
Inadvertent soil ingestion (milligrams[c] per day)	50
Inhalation exposure time (hours)	5,073.5
Soil exposure time (hours)	2,387

a. To convert kilograms to pounds, multiply by 2.2046.
b. To convert liters to gallons, multiply by 0.26417.
c. To convert milligrams to ounces, multiply by 0.000035274.

The biosphere dose conversion factors for the RMEI, characterized by the set of attributes listed in Table I-10, are given in Table I-11.

I.4.4.7 Igneous Activity Scenario

The model and parameter changes from TSPA–Site Recommendation to the model configuration used in the analysis for this EIS for the igneous activity scenario are described in detail in *FY01 Supplemental Science and Performance Analyses*, Volume 1 (DIRS 155950-BSC 2001, Sections 13 and 14) and are summarized here.

Several input parameters to the TSPA models used to calculate consequences of igneous disruption changed after the TSPA–Site Recommendation and have been included in this analysis (DIRS 155950-

Table I-11. Biosphere dose conversion factors for the RMEI for the groundwater release and the volcanic release exposure scenarios.

Radionuclide	Groundwater release[a] (rem per picocurie per liter[b])	Volcanic release[a] (rem per picocurie per square meter[c])
Carbon-14	0.000029	NA[d]
Selenium-79	0.000012	3.8×10^{-11}
Strontium-90	0.0002	4.2×10^{-9}
Technetium-99	0.0000028	NA
Iodine-129	0.00025	NA
Cesium-137	0.00034	1.2×10^{-9}
Lead-210	0.0051	1.4×10^{-8}
Radium-226	0.005	4.2×10^{-9}
Actinium-227	0.013	1.9×10^{-6}
Thorium-229	0.0061	6.0×10^{-7}
Thorium-230	0.0012	9.1×10^{-8}
Protactinium-231	0.016	3.8×10^{-7}
Uranium-232	0.0018	1.9×10^{-7}
Uranium-233	0.00028	3.8×10^{-8}
Uranium-234	0.00027	3.8×10^{-8}
Uranium-236	0.00026	NA
Uranium-238	0.00026	NA
Neptunium-237	0.0045	1.9×10^{-7}
Plutonium -238	0.0029	1.1×10^{-7}
Plutonium-239	0.0035	1.3×10^{-7}
Plutonium -240	0.0035	1.3×10^{-7}
Plutonium-242	0.0032	1.2×10^{-7}
Americium-241	0.0035	1.3×10^{-7}
Americium-243	0.004	1.3×10^{-7}

a. Biosphere Dose Conversion Factors for the transition phase, 1 centimeter (0.4 inch) layer of ash and annual average mass loading
b. To convert liters to gallons, multiply by 0.26417.
c. To convert from square meters to square feet, multiply by 10.764.
d. NA = not applicable.

BSC 2001, Section 14.3.3.7). Consistent with new information regarding the probability of an eruption at the location of the proposed repository given an igneous intrusive event (DIRS 155950-BSC 2001, Section 14.3, all), the conditional probability of an eruption at the proposed repository was revised from 0.36 (DIRS 153246-CRWMS M&O 2000, Table 3.10-4, p. 198) to 0.77 (DIRS 155950-BSC 2001, Section 14.3.3.1, p. 14-13). According to *Characterize Framework for Igneous Activity at Yucca Mountain, Nevada* (DIRS 151551-CRWMS M&O 2000, Section 6.5.3.2, and Table 12a, p. 130), the approach for the calculation of the conditional number of eruptive centers occurring within the repository footprint was modified by: 1) using empirical distributions for the average spacing between eruptive centers rather than the expected values for these distributions, and 2) incorporating uncertainty in the effect of the repository opening on the conditional probability of the occurrence of an eruptive center within the repository footprint. This modified approach resulted in the new conditional probability of 0.77 for one eruptive center to occur involving the Primary Block of the higher-temperature repository operating mode footprint during or coincident with an igneous activity event. This conditional probability has also been assumed for the lower-temperature operating mode analyses in Section I.5.

Changes also were made in the probability distribution for an intrusive event, consistent with revisions in the repository footprint (changes related to the higher-temperature operating mode) because inputs were compiled for TSPA–Site Recommendation. Revised distributions were provided for the number of waste packages affected by igneous intrusion and volcanic eruption events, consistent with the revised event probability information for the Primary Block of the higher-temperature operating mode. This adjusted event probability has also been assumed for the lower-temperature operating mode analyses in Section I.5. Changes have been made in the input data used to determine the wind speed during an eruption (DIRS 155950-BSC 2001, Section 3.3.1.2.1). Additional changes in inputs to the TSPA–Site Recommendation igneous consequence model are listed in *FY01 Supplemental Science and Performance Analyses*, Volume 1 (DIRS 155950-BSC 2001, Section 14.3.3.7, p. 14-24, and Tables 14.3.3.7-1 and 14.3.3.7-2, p. 14T-5 to 14T-6). Other model inputs and assumptions, including the assumption that wind direction would be fixed toward the location of the exposed individual at all times, were the same as those used in the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, Section 3.10).

I.4.4.8 Human Intrusion Scenario

The human intrusion scenario for the calculations in this EIS was developed from that in the TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, Section 4.4). The model changes implemented for the human intrusion calculations in this EIS are described in this section.

Errata in the TSPA–Site Recommendation human intrusion model associated with “boosting” the inventory of certain radionuclides to account for first-generation decay product transport through the three-dimensional saturated zone model (DIRS 148384-CRWMS M&O 2000, Section 6.3.4.1, p. 233) were corrected.

In the TSPA–Site Recommendation human intrusion submodel (DIRS 148384-CRWMS M&O 2000, Section 6.3.9.3, p. 513), for the purpose of determining thermal-hydrologic conditions, in-package chemistry, and in-drift chemistry, the failed waste package was placed in a specified dripping environment for a given infiltration condition. For the calculations in this EIS, the failed waste package for each realization of the human intrusion scenario was randomly placed in one of several dripping environments depending on the infiltration condition.

Colloidal-facilitated transport of americium, plutonium, thorium, and protactinium in an exploratory borehole through the unsaturated zone has been included in the human intrusion scenario in this EIS. The decay products of irreversibly sorbed americium-241 and neptunium-237 were included as an irreversibly sorbed colloidal species. Colloidal-facilitated transport was implemented by adjusting the sorption coefficients of the aforementioned nuclides according to the relationship (DIRS 139440-CRWMS M&O 2000, p. 26):

$$K_d^{adj} = \frac{K_d^{orig}}{1 + K_c}$$

where

K_d^{orig} = sorption coefficient without colloidal-facilitated transport

K_d^{adj} = sorption coefficient with colloidal-facilitated transport

K_c = colloid partition coefficient

The human intrusion scenario in this EIS was simulated for a 1-million-year duration (as opposed to the 100,000-year duration in the TSPA–Site Recommendation). To be consistent with the *FY01 Supplemental Science and Performance Analyses* 1-million-year calculations, two additional radionuclides, radon-228 and thorium-232, were included in the inventory (DIRS 155950-BSC 2001, Section 13.2.1.10, pp. 13-9 and 13-10). The 30,000-year human intrusion scenario is the same scenario analyzed in the *FY01 Supplemental Science and Performance Analyses* (DIRS 155950-BSC 2001, Volume 1, Appendix A). The information in that appendix addresses the issue of when a human intrusion could occur based upon the earliest time that current technology and practices could lead to waste package penetration without the driller noticing waste package penetration. The earliest time would be that time (approximately 30,000 years) when the waste package had corroded sufficiently that a drill bit could penetrate it.

The assessment of the human intrusion scenario did not combine the results of this scenario with the results of the disruptive igneous activity scenario. However, combined results can be approximated by adding the results of the human intrusion calculations to the results of the disruptive igneous event scenario. Based on the results in Section I.5, the highest mean annual individual dose that would result from a human intrusion would be less than one-tenth of the radiological dose from a disruptive igneous event.

I.4.5 EXTENSION OF GROUNDWATER IMPACTS TO OTHER DISTANCES

The TSPA model described in Section I.2 was used to model the environmental impacts to groundwater for the long-term postclosure period. The TSPA model was originally developed to support the Yucca Mountain site suitability evaluation and possible subsequent licensing compliance analyses for the repository if the site was recommended. The model is, therefore, focused on the compliance requirements set forth in applicable regulations such as the Environmental Protection Agency standard, 40 CFR Part 197. This standard is concerned with a single compliance point, the RMEI location. The long-term impacts to groundwater predicted by the TSPA model would be restricted to that single location. Supporting models, such as the site-scale flow and transport model, were developed to support the TSPA calculation and do not extend much beyond the RMEI location. Furthermore, the TSPA made a conservative assumption that all radionuclide mass in groundwater would be captured in the water usage at the RMEI location. This is a reasonable approach for the compliance calculations because it tends to bias the concentration of materials to a higher value, without trying to account for complicated plume-capture considerations, and also because the volume of the plume passing this point in 1 year would be on the order of the upper bound of water usage. However, this assumption in the model does not allow it to account for the spreading of the plume at greater distances considered in this EIS.

As part of a comprehensive presentation of impacts, this EIS is charged with providing groundwater impacts for two other important downgradient locations. These are 30 kilometers (18 miles), where most of the current population in the groundwater path is located, and 60 kilometers (37 miles), where the aquifer discharges to the surface (also known as Franklin Lake Playa). The selection of these locations is discussed in Section I.4.5.1.

To provide insight about impacts at these other distances, a method of scaling was developed. This was necessary because the TSPA model is limited to the RMEI location, as described above. This section describes the approach to the scaling and the results obtained. The scaling approach is discussed in Section I.4.5.2 and the scaling factors in Section I.4.5.3.

I.4.5.1 Locations for Assessing Postclosure Impacts to Human Health

The Environmental Protection Agency public health and environmental radiation protection standards for Yucca Mountain (40 CFR Part 197) require DOE to estimate the potential radiation doses to the public from the disposal of spent nuclear fuel and high-level radioactive waste, based on the concept of the RMEI. This involves estimating the dose to a person assumed to be among those at greatest risk for 10,000 years after repository closure, given certain conservative exposure parameters and parameter value ranges. The Environmental Protection Agency selected a theoretical individual representative of a future population group or community, termed *rural-residential*, as the basis of an individual exposure scenario. This rural-residential RMEI would be exposed through the same general pathways as a subsistence farmer; however, the RMEI would not be a full-time farmer but rather would consume some locally grown food (self-grown or from local sources) as part of the exposure scenario. The Environmental Protection Agency also established a maximum 300-square-kilometer (74,000-acre)-controlled area, and established a RMEI location that equates to approximately 18 kilometers (11 miles) south of the repository (the predominant direction of groundwater flow), for demonstrating compliance with the long-term performance standards. The Environmental Protection Agency standard defines the postclosure accessible environment as being any point outside the controlled area.

For purposes of estimating potential environmental impacts in this EIS, DOE considered the impacts to the RMEI approximately 18 kilometers (11 miles) downgradient from the repository, as well as at other reasonable locations. In determining those locations, DOE considered locations where it would be reasonable from a technical and economic standpoint to locate a rural-residential individual. Although there exists a large number of locations at which analyses could be performed, DOE has determined that the most reasonable analyses to perform are for a rural-residential individual approximately 18, 30, and 60 kilometers (11, 19, and 37 miles) downgradient from the proposed repository, because these locations are based on realistic exposure conditions that would provide the basis for a meaningful comparison of potential human health impacts.

The Environmental Protection Agency, in reaching its conclusion on the location of the southernmost extent of the controlled area, considered current and projected uses of the land in the vicinity of the area formerly known as Lathrop Wells [now known as Amargosa Valley, approximately 20 kilometers (12 miles) downgradient from the repository]. The Agency noted there are currently eight residents and fewer than 10 businesses near this location whose source of water is the aquifer that flows beneath Yucca Mountain. This is the location where private property is nearest the proposed repository, and where some soils are suitable for agricultural purposes [the nearest farm is somewhat farther south, about 23 kilometers (14 miles) downgradient from the repository]. The Agency used near-term projections of land development between the current population at Amargosa Valley north to the Nevada Test Site. Near-term plans for the area between Amargosa Valley and the Test Site boundary include a science museum and industrial activities. Therefore, the boundary of the Test Site was used as the southernmost extent of the controlled area in 40 CFR 197.12. For this EIS, DOE adopted the southernmost extent of the controlled area as the RMEI location. This location is about 18 kilometers (11 miles) downgradient from the proposed repository.

DOE also has identified other reasonable locations for a hypothetical rural-residential individual approximately 30 and 60 kilometers (19 and 37 miles) downgradient from the repository. The closest population center is 30 kilometers (19 miles) downgradient in Amargosa Valley. At this location, the depth to groundwater suitable for human consumption and other uses (for example, agricultural) ranges from about 9 to 40 meters (30 to 130 feet) deep (less than that at the location formerly known as Lathrop Wells), and wells supply water to individual households. Franklin Lake Playa is about 60 kilometers (37 miles) downgradient from the proposed repository and is the location where the aquifer could emerge as surface water.

In conclusion, these three locations where a rural-residential individual could be reasonably located [about 18, 30, and 60 kilometers (11, 19, and 37 miles) downgradient] represent realistic locations where water for human consumption and other uses can occur using commonly available techniques without undue costs to withdraw and distribute water. These locations also reflect current populations and lifestyles in areas where dissolved radionuclides in the groundwater could affect future populations.

In the Draft EIS, DOE analyzed a Maximally Exposed Individual at a location 5 kilometers (3 miles) from the repository. The Maximally Exposed Individual was defined as a hypothetical person exposed to radiation in such a way—by a combination of factors including location, lifestyle, dietary habits, and so on—that the individual would be the most highly exposed member of the public. The Maximally Exposed Individual in the Draft EIS was a hypothetical member of a group of adults that would live in the Amargosa Valley with a characteristic range of lifestyle, food consumption, and groundwater usage patterns. This individual would grow half of the foods that the individual would consume on the property, irrigate crops and water livestock using groundwater, and use groundwater as a drinking water source and to bathe and wash clothes. The lifestyle and related exposure characteristics of the Maximally Exposed Individual are similar to those of the Environmental Protection Agency's rural-residential RMEI.

DOE noted in the Draft EIS that there are no permanent residents at a location 5 kilometers (3 miles) downgradient from the repository. The water table lies more than 360 meters (1,200 feet) deep in hard, volcanic rock. Although it might be possible, DOE would not expect permanent residents at that location in the future because of a lack of economically accessible groundwater. Human habitation has occurred in the vicinity of the repository where only the groundwater is easily accessible. Furthermore, the lands in this area are under the control of the Federal Government and within the controlled area defined in 40 CFR Part 197 – and thus are not part of the postclosure accessible environment.

In spite of these factors, DOE analyzed a Maximally Exposed Individual at a location 5 kilometers (3 miles) downgradient of the repository in the Draft EIS. At the time of the Draft EIS, Environmental Protection Agency had not published its draft or final radiation protection standards for Yucca Mountain, but DOE believed that a 5-kilometer compliance location could be established by the Environmental Protection Agency, given a similar compliance location in its generally applicable standards for the disposal of spent nuclear fuel, high-level radioactive waste and transuranic waste (40 CFR 191).

However, the Environmental Protection Agency has since published its final Yucca Mountain-specific public health and environmental radiation protection standards, and has concluded:

> "…it improbable that the rural-residential RMEI [reasonably maximally exposed individual] would occupy locations significantly north of U.S. Route 95 [location formerly known as Lathrop Wells], because the rough terrain and increasing depth to ground water nearer Yucca Mountain would likely discourage settlement by individuals because access to water is more difficult than it would be a few kilometers farther south."

The Environmental Protection Agency considered whether or not the inherent nature of the soils and the topography were conducive to or would constrain further development of the area near Yucca Mountain. The Agency concluded that:

> "…agricultural activity would be limited around Yucca Mountain as a result of adverse conditions, such as steep slopes, rocky terrain, and shallow soils…"

The Environmental Protection Agency also considered the potential dose to a RMEI at locations closer than approximately 18 kilometers (11 miles), and concluded that a rural-residential individual would receive a lower dose than those at 18 kilometers. The Agency stated that:

> "If individuals lived near the repository, they would be unlikely to withdraw water from the significantly greater depth for other than domestic use, and in the much larger quantities needed for gardening or farming activities because of the significant cost of finding and withdrawing the ground water. It is possible, therefore, for an individual located closer to the repository to incur exposures from contaminated drinking water, but not from ingestion of contaminated food. Based upon our analyses…we believe that use of contaminated ground water…would be the most likely pathway for most of the dose from the most soluble, more mobile radionuclides…The percentage of the dose that results from irrigation would depend upon assumptions about the fraction of all food consumed by the RMEI from gardening or other crops grown using contaminated water, which should reflect the

> lifestyle of current residents of the Town of Amargosa Valley. Therefore, the exposure of an RMEI located approximately 18 km south of the repository…actually would be more conservative than an RMEI located much closer to the repository…"

The Agency also addressed the economic feasibility of well drilling and pumping costs and concluded that:

> "…the capital costs of private wells for domestic use become prohibitive at depths between 300 and 600 feet. For communal domestic use and irrigation use, the capital costs do not become prohibitive even at depths of 1,200 feet…However, because of the very large volumes of water needed for irrigating field crops, particularly in the climate of Amargosa Valley, pumping costs are very significant for such agricultural applications. Combining the pumping cost estimates…with the capital cost estimates…the marginal value of water for irrigation is exceeded at depths to water greater than 300 feet. In fact, since these estimates do not consider the distribution cost for the irrigation system or any maintenance costs…it is not surprising to see that commercial agricultural activities in Amargosa Valley have been restricted thus far to areas where the depth to water is generally less than about 200 feet."

Based on the above considerations, DOE did not reevaluate the impacts at 5 kilometers (3 miles). This EIS contains evaluations of impacts at the RMEI location, at 30 kilometers (19 miles) downgradient from the repository (population center), and at the groundwater surface discharge point 60 kilometers (37 miles) downgradient from the repository.

I.4.5.2 Scaling Approach

This section summarizes the approach detailed in DIRS 157520-Williams (2001, Enclosure 3).

As the plume traveled over a given distance in the saturated zone, the concentration of radionuclides in the plume could be attenuated by several effects: dispersion, decay, filtration of solids by the aquifer medium, irreversible sorption of radionuclides by the aquifer medium, and other minor phenomena. The dispersion effects would be due to the combination of molecular diffusion and hydrodynamic mixing, that would tend to cause the contaminants to spread out along and transverse to the path of flow. The dispersion effect would reduce the peak concentration of the plume and increase the volume of the plume. The decay effect would be due to the later arrival of the plume centerline at a farther distance, allowing time for nuclear decay. The travel time would depend on the flow rate of the water and the retardation of contaminants that were sorbed reversibly by the aquifer solid media. The overall reduction by decay would be governed by the radionuclide travel time and the rate of decay of a particular radionuclide. The effects of colloid filtration, irreversible sorption, and other minor phenomena are expected to be small and are normally neglected. The principal radionuclides that would contribute to dose and most significantly affect groundwater quality have very long half-lives (and therefore very slow rates of decay), so the reduction of concentration by decay would be fairly small. The major contributor to the reduction of concentration in the contaminant plume, then, would be the dispersion effect. Therefore, the scaling approach was developed from only the dispersion effect. This produced a conservative result because the decay effect will cause some small additional reduction in concentration.

All of the major attenuating effects listed above were applied in the TSPA model for the calculation of the dose and water quality at the compliance point. However, because most of the path from the proposed repository to the compliance point is in the volcanic aquifer, there is only a small amount of dispersion. The volcanic aquifer is comprised mostly of fractured rock, so flow occurs in small isolated channels and mixing is minimal. This is why the plume is still small at the compliance point and full capture is a reasonable assumption. In the alluvial aquifer that extends from the RMEI location down to the discharge point, the aquifer medium is a finely divided, granular material where flow is slow and considerably more mixing can occur.

An analytical solution to the three-dimensional advection-dispersion problem was used to estimate dispersion effects from the RMEI location to the discharge point (DIRS 157520-Williams 2001, Enclosure 3, all). In these calculations, the groundwater flow velocity in the alluvium was assumed to be horizontal with a constant value of 18 meters (59 feet) per year, corresponding to a specific discharge rate of 2.7 meters (9 feet) per year and an effective porosity of 15 percent throughout the flow domain. These values were derived from the saturated zone site-scale model documented in DIRS 155950-BSC (2001, Section 12). Calculations were done under steady-state conditions, that is, for a source that has been discharging for a long time. The source was assumed to have constant concentration, be within a rectangular shape in the vertical plane, and centered at the repository location. Two source sizes were considered: a small source, 10 meters by 10 meters (33 feet by 33 feet), corresponding to an early failure scenario (localized failing waste package), and a large source, 3,000 meters (9,840 feet) horizontal by 10 meters (32.8 feet) vertical, corresponding to a long-term scenario in which all waste packages would fail.

The calculations were carried out for a range of dispersivities and for two assumed mass captures: 90 percent and 99 percent. The mass capture is a function of the amount of influence a well or field of wells would have in pulling mass from the plume. The results discussed here are restricted to the more conservative 99-percent capture assumption. Two important parameters were considered: the cross-section (perpendicular to flow) of the plume and the relative peak concentration at the center of the plume. As the plume traveled in the groundwater it would spread, so the cross-section would increase (thus reducing the average concentration) and the peak concentration would decrease. A reasonable approximation of distance effect can then be found by using either of these values. The two values will produce a slightly different result. Scaling factors using both approaches are discussed in the next section.

I.4.5.3 Scaling Factors for Dose or Water Quality Concentrations at Longer Distances

Table I-12 lists the resulting scaling factors from the dispersion studies (DIRS 157520-Williams 2001, Enclosure 3, Table 2a). The values are for the assumption of 99-percent capture, the larger realistic dispersion factor set, and two source sizes. The large source size would be applied for nominal scenario peak dose and the small source for localized sources such as the early failures (prior to 10,000 years) due to package defects or igneous intrusion releases, or for doses from the human intrusion scenario. Two sets of scaling factors are listed for each source size: one based on peak concentration and one based on plume cross-section. To obtain a value of dose or groundwater quality concentration at 30 or 60 kilometers (18 or 37 miles), multiply the 18-kilometer (11-mile) value by the appropriate scaling factor. The scaled results reported in Chapter 5, Section 5.4.1, use the plume cross-section factors. This is considered the best choice because the effect of water usage by the communities would be to cause significant mixing, and the more characteristic parameter would be the plume average concentration.

I.5 Waterborne Radioactive Material Impacts

The simulations in support of this analysis estimated the annual individual dose for the Proposed Action, Module 1, and Module 2 inventories. For the purposes of this EIS, DOE determined that the southern boundary of the controlled area would be at the southernmost point from the repository specified in 40 CFR Part 197 (36 degrees, 40 minutes, 13.6661 seconds north latitude). The RMEI location was then defined to be the point where the predominate groundwater flow crosses the boundary. Groundwater modeling indicated this point to be approximately 18 kilometers (11 miles) downgradient from the potential repository. This EIS refers to this location as the “RMEI location.” It corresponds to where the RMEI, a resident in an average farming community, would consume and use groundwater withdrawn from wells. In accordance with 40 CFR 197.35, the annual individual dose was calculated for the period of geologic stability (1 million years). These calculations include simulations for both the 10,000- and 1 million-year performance periods specified in 40 CFR 197.20 and 197.35.

Table I-12. Groundwater impact distance scale factors[a,b] for 99-percent captured mass, longitudinal dispersivity 100 meters,[c] horizontal dispersivity 10 meters, and vertical dispersivity 0.1 meters.

Source	Scale factors	
	18 kilometers[d] to 30 kilometers	18 kilometers to 60 kilometers
Large source: 3,000 × 10 meters		
Based on plume cross-section	0.68	0.39
Based on relative peak concentration	0.74	0.46
Small source: 10 × 10 meters		
Based on plume cross-section	0.70	0.48
Based on relative peak concentration	0.60	0.30

a. Derived from DIRS 157520-Williams (2001, Enclosure 3, Table 2a).
b. To convert an 18-kilometer result to a 30- or 60-kilometer result, multiply the dose or the concentration by the appropriate value in the table.
c. To convert meters to feet, multiply by 3.281.
d. To convert kilometers to miles, multiply by 0.6214.

The calculations in this EIS also show the peak dose for all scenarios. The location is also where a representative volume of groundwater would be withdrawn and where there would be a reasonable expectation that radiation would not exceed the limits of 40 CFR 197.30, Table 1. This EIS also reports groundwater protection values at that location.

The data from the multiple realizations can be summarized by showing time versus annual individual dose (dose histories) for the 5th-percentile, median, mean, and 95th-percentile of the output. In the manner described for TSPA–Site Recommendation (DIRS 153246-CRWMS M&O 2000, Section 2.2.4.6, pp. 2-39 to 2-40), these statistical measures were calculated for all 300 realizations of the probabilistic simulations at each time step of the annual individual dose histories. The plot of the mean represents the average of all 300 data points at each time step. For each point on the plot of the median dose, 50 percent of the data have a value greater than the plotted point and 50 percent have a value less than the plotted point. Similarly, for the 5th- and 95th-percentiles, the plotted data points are such that 95 percent of data are greater than the plotted point and 5 percent of the data points are greater than the plotted points, respectively, for each time step. The statistical measures were superimposed on plots that show all 300 realizations (often referred to as "horsetail plots").

I.5.1 WASTE PACKAGE FAILURE

Figure I-8 shows the waste package failure curves for the Proposed Action for the 1-million-year performance period for the higher-temperature operating mode. The figure indicates that the first waste package failures would occur within 10,000 years of repository closure. These early waste package failures result from the assumption of improper heat treatment (see Section I.2.4 and Table I-3). The 300 realizations are shown in Figure I-8. During the first 10,000 years there are some realizations showing a failure fraction of 0.00025, which when multiplied times the total waste packages (11,770) gives a maximum of 3 early waste package failures. There are some realizations that show zero failures, but this is not readily evident from the figure. Waste package failure would be the first step in releasing radionuclides for groundwater flow and transport.

Figure I-9 shows cladding perforated during the postclosure period. The calculations included the averaged impact of seismic events. The cladding failure results shown in Figure I-9 are essentially the same as those developed in the *FY01 Supplemental Science and Performance Analyses* (DIRS 154659-BSC 2001, pp. 9-19 to 9-23).

I.5.2 ANNUAL INDIVIDUAL DOSE FOR 10,000 YEARS AFTER CLOSURE

This section presents graphic representations of annual individual doses for the inventories described in Section I.3 and the scenarios described in Section I.4. The performance period for the calculations in this EIS was generally 1 million years after repository closure except in the case of the igneous activity scenarios. The annual dose histories for the igneous activity scenarios were only calculated for 100,000 years after closure because the releases from the nominal scenario dominate after that time. In addition to the graphic presentations, Table I-13 lists the values of the peak mean annual individual dose for all scenarios that would occur in the 10,000-, 100,000-, and 1-million-year postclosure performance periods, in accordance with 40 CFR 197.20, 197.25, and 197.35. Table I-14 lists the same information for the peak 95th-percentile annual individual dose.

Table I-13. Peak mean annual individual doses (millirem) for analyzed inventories, scenarios, and temperature operating modes.[a,b]

Modeled inventory, scenario, and operating mode	10,000 years		100,000 years		1 million years	
	Value	Year	Value	Year	Value	Year
Proposed Action, nominal, higher-temperature	0.000017	4,875	0.12	99,500	152.5	476,000
Proposed Action, nominal lower-temperature	0.000011	3,437.5	0.085	99,500	122.2	476,000
Inventory Module 1, nominal, higher-temperature	0.000027	4,937.5	0.16	100,000	237.9	476,000
Inventory Module 2, nominal, higher-temperature[c]	0.00066	2,875	0.00066	2,875	0.33	208,000
Proposed Action, igneous activity, higher-temperature	0.10	312.5	0.10	312.5	NC[d]	NC
Proposed Action, igneous activity, lower-temperature	0.10	312.5	0.10	312.5	NC	NC
Proposed Action, igneous activity (intrusive only), higher-temperature	0.00043	10,000	0.021	48,000	NC	NC
Proposed Action, igneous activity (intrusive only), lower-temperature	0.00050	10,000	0.028	48,000	NC	NC
Proposed Action, igneous activity (eruptive only), higher-temperature	0.10	312.5	0.10	312.5	NC	NC
Proposed Action, igneous activity (eruptive only), lower-temperature	0.10	312.5	0.10	312.5	NC	NC
Proposed Action, human intrusion at 30,000 years, higher-temperature	NA[e]	NA	0.0017	30,562.5	0.0023	108,000

a. Adapted from DIRS 157307-BSC (2001, Enclosure 1).
b. These data are based on the same probabilistic annual water usage model used in the TSPA–Site Recommendation (not 3,000 acre-feet per year).
c. Module 2 runs only included the incremental effect of the additional inventory from Greater-Than-Class-C and Special-Performance-Assessment-Required waste.
d. NC = not calculated.
e. NA = not applicable.

I.5.3 ANNUAL INDIVIDUAL DOSE FOR 1,000,000 YEARS AFTER CLOSURE

Results for annual individual dose calculations for 1 million years following closure are discussed for the Proposed Action (Section I.5.3.1), Inventory Module 1 (Section I.5.3.2) and Inventory Module 2 (Section I.5.3.3).

Table I-14. Peak 95th-percentile annual individual doses (millirem) for analyzed inventories, scenarios, and temperature operating modes.[a,b]

Modeled inventory, scenario, and operating mode	10,000 years		100,000 years		1,000,000 years	
	Value	Year	Value	Year	Value	Year
Proposed Action, nominal, higher-temperature	0.00012	4,937.5	0.040	99,500	618.0	408,000
Proposed Action, nominal, lower-temperature	0.000086	5,000	0.034	100,000	513.2	408,000
Inventory Module 1, nominal, higher-temperature	0.00018	4,125	0.079	100,000	976.7	476,000
Inventory Module 2, nominal, higher-temperature[c]	0[d]	NA[e]	0.0013	100,000	1.5	208,000
Proposed Action, igneous activity, higher-temperature	0.41	312.5	0.41	312.5	NC[f]	NC
Proposed Action, igneous activity, lower-temperature	0.41	312.5	0.41	312.5	NC	NC
Proposed Action, igneous activity (intrusive only), higher-temperature	0.00029	9,750	0.052	100,000	NC	NC
Proposed Action, igneous activity (intrusive only), lower-temperature	0.00031	9,875	0.033	48,000	NC	NC
Proposed Action, igneous activity (eruptive only), higher-temperature	0.41	312.5	0.41	312.5	NC	NC
Proposed Action, igneous activity (eruptive only), lower-temperature	0.41	312.5	0.41	312.5	NC	NC
Proposed Action, human intrusion at 30,000 years, higher-temperature	NA	NA	0.0045	38,500	0.0045	38,400

a. Adapted from DIRS 157307-BSC 2001, Enclosure 1.
b. These data are based on the same probabilistic annual water usage model used in the TSPA–Site Recommendation (not 3,000 acre-feet per year).
c. Module 2 runs only included the incremental effect of the additional inventory from Greater-Than-Class-C and Special-Performance-Assessment-Required waste.
d. The mean dose is driven by 3 realizations that experience early failures; no other realizations result in a dose before 10,000 years so that the 95th-percentile value is zero.
e. NA = not applicable.
f. NC = not calculated.

I.5.3.1 Annual Individual Dose for the Proposed Action Inventory, Higher- and Lower-Temperature Repository Operating Modes

Figure I-10 shows the mean annual individual dose results of the 300 probabilistic simulations for the higher-temperature repository operating mode (approximately 56 MTHM per acre) for the Proposed Action inventory at the RMEI location for 1 million years after repository closure. Figure I-11 shows the relative contribution of selected radionuclides that contribute most to the total mean annual dose due to all radionuclides. Figure I-12 shows the results of the 300 probabilistic simulations of the Proposed Action inventory, higher-temperature operating mode, at the RMEI location for 1 million years after repository closure. This figure shows the results for each realization and the 5th-percentile, mean, median, and 95th-percentile of these simulations.

Figure I-10 also shows representations of the mean annual individual dose results of the 300 probabilistic simulations for the lower-temperature operating mode (approximately 45 MTHM per acre) for the Proposed Action inventory at the RMEI location for 1 million years after repository closure. Because Figure I-10 shows little difference between the annual individual dose histories calculated for the higher-temperature and the lower-temperature operating modes, the remaining scenarios, other than the igneous activity scenario, were simulated only for the higher-temperature operating mode. Figure I-13 shows the results of the 300 probabilistic simulations of the Proposed Action inventory, lower-temperature operating mode, at the RMEI location for 1 million years after repository closure. This figure shows the results for each realization and the 5th-percentile, mean, median, and 95th-percentile of these simulations.

I.5.3.2 Annual Individual Dose for Inventory Module 1, Higher-Temperature Repository Operating Mode

Figure I-14 displays the annual dose histories for the 300 probabilistic simulations of the expanded-inventory Module 1, higher-temperature operating mode at the RMEI location for 1 million years after repository closure. This figure shows the results for each realization and the 5th-percentile, mean, median, and 95th-percentile of these simulations.

I.5.3.3 Annual Individual Dose for Inventory Module 2, Higher-Temperature Repository Operating Mode

A GoldSim simulation was performed for a case that included only the Greater-Than-Class-C and Special-Performance-Assessment-Required components of the Module 2 inventory. The case did not include the other components of the Module 2 inventory (that is, the Module 1 inventory). The GoldSim simulation for only the Module 2 Greater-Than-Class-C and Special-Performance-Assessment-Required inventory, higher-temperature operating mode, was performed as a separate probabilistic case at the RMEI location. Figure I-15 shows the results of this simulation as the mean annual individual dose due to the radioactive components of this material. The effects of nonradioactive components of this waste are not included in the analysis.

Figure I-16 is a comparison plot of the mean annual dose versus time for the Proposed Action, Module 1, and the Greater-Than-Class-C and Special-Performance-Assessment-Required waste portion of the Module 2 inventories at the higher-temperature operating mode at the RMEI location. These results show that during the first 10,000 years, the mean annual individual dose due to the Greater-Than-Class-C and Special-Performance-Assessment-Required components of the Module 2 inventory would be greater than that calculated for the Proposed Action and Module 1 inventories, but still essentially zero. After 10,000 years, the dose due to the Greater-Than-Class-C and Special-Performance-Assessment-Required components of the Module 2 inventory would be about two orders of magnitude less than that calculated for the Proposed Action and Module 1 inventories. These results indicate that the addition of the Greater-Than-Class-C and Special-Performance-Assessment-Required waste to the Module 1 inventory would not materially increase the mean annual individual dose. Based on this comparison, separate probabilistic simulations were not run for the entire Inventory Module 2.

I.5.3.4 Annual Individual Dose for Igneous Activity Scenario, Higher- and Lower-Temperature Repository Operating Modes

The performance of a Yucca Mountain repository was evaluated for a combined igneous activity scenario that included both an igneous event and a volcanic eruption. The combined scenario was simulated for the higher- and lower-temperature repository operating modes for the Proposed Action inventory. Annual dose histories were not calculated for the igneous activity scenario for Modules 1 and 2.

Figure I-17 shows representations of the probability-weighted annual individual dose histories for 500 of the 5,000 probabilistic simulations for the igneous activity scenario, higher-temperature repository operating mode (approximately 56 MTHM per acre) for the Proposed Action inventory at the RMEI location for 100,000 years after repository closure. Figure I-17 also shows the 5th-percentile, mean, median, and 95th-percentile of all 5,000 igneous activity simulations. The results shown in the figure represent the combined effect of both the igneous-intrusion and eruptive events.

Figure I-18 shows the mean annual individual dose versus time for the igneous activity scenario for the Proposed Action inventory for the higher-temperature operating mode at the RMEI location. The figure also shows the mean results for both the eruptive and intrusive events. Figure I-19 shows the mean annual individual dose for the igneous activity scenarios, representing the sum of the igneous and eruptive events, Proposed Action inventory for the higher- and lower-temperature operating modes at the RMEI location.

Figure I-20 shows representations of the probability-weighted annual individual dose histories for 500 of the 5,000 probabilistic simulations for the igneous activity scenario, lower-temperature repository operating mode (approximately 45 MTHM per acre) for the Proposed Action inventory at the RMEI location for 100,000 years after repository closure. Figure I-20 also shows the 5th-percentile, mean, median, and 95th-percentile of all 5,000 igneous activity simulations. The results presented in this figure represent the combined effect of both the igneous intrusion and eruptive events.

Figure I-21 shows the mean individual annual dose versus time for the igneous activity scenario for the Proposed Action inventory for the lower-temperature repository operating mode at the RMEI location. The figure also displays the mean results for both the eruptive and intrusive events.

I.5.3.5 Annual Individual Dose for the Human Intrusion Scenario

Figure I-22 displays representations of the annual individual dose results of the 300 probabilistic simulations for the human intrusion scenario, 30,000 years after repository closure, Proposed Action inventory for the higher-temperature operating mode at the RMEI location. Figure I-22 displays the results for each simulation and the 5th-percentile, median, mean, and 95th-percentile of these simulations.

I.5.4 COMPARISON TO GROUNDWATER PROTECTION STANDARDS

An analysis for groundwater protection was conducted in accordance with the Environmental Protection Agency Final Rule 40 CFR 197.30 and 197.31). The rule is based on meeting three groundwater radionuclide-concentration levels. The first is the maximum annual concentration of radium-226 and -228 in a representative volume of 3.7 million cubic meters (3,000 acre-feet) of groundwater in a release from the proposed repository. The second groundwater concentration is for the gross alpha activity (excluding radon and uranium) in the representative volume of groundwater. Both calculations apply to releases from both natural sources and releases from the repository at the same location as the RMEI. The third groundwater-protection calculation is the dose to the whole body or any organ of a human for beta- and photon-emitting radionuclides released from the repository. The human would consume 2.0 liters (0.53 gallon) per day from the representative volume of groundwater. This groundwater would be withdrawn annually from an aquifer containing less than 10,000 milligrams per liter (1.34 ounces per gallon) of total dissolved solids, and centered on the highest concentration in the plume of contamination at the same location as the RMEI. The results of the calculations for this EIS produced data consistent with the Environmental Protection Agency Final Rule and are presented graphically and in tabular form.

Figure I-23 shows the mean activity concentrations of gross alpha activity and total radium (radium-226 plus radium-228) in the representative volume of groundwater for the Proposed Action inventory, higher-temperature repository operating mode. The concentrations are calculated for a representative volume of water of 3.7 million cubic meters (exactly 3,000 acre-feet per year) at the same location as the RMEI at the accessible environment as described in 40 CFR 197.30. Naturally occurring background radionuclide concentrations were not included because the calculated values are negligible compared to background concentrations up to 100,000 years after closure. Figure I-24 shows the same information for the lower-temperature operating mode.

Figure I-25 shows the mean dose to the whole body or any organ for technetium-99, carbon-14, and iodine-129, the prominent beta and photon-emitting radionuclides (DIRS 154659-BSC 2001, Volume 2, Section 4.1.4, pp. 4 to 11) for the Proposed Action inventory, higher-temperature repository operating mode, for the 1-million-year performance period. Figure I-26 shows the same information for the lower-temperature operating mode.

The data developed for the groundwater protection standard are summarized in Table I-15, which lists the peak mean gross alpha activity by scenario for various performance periods; Table I-16, which lists peak total radium concentration by scenario for various performance periods; and Table I-17, which lists the combined whole-body or organ doses in 10,000 years for the total of all beta- and photon-emitting radionuclides. The mean whole-body or organ dose was calculated by diluting the model-predicted annual activity releases of iodine-129, carbon-14, and technetium-99 [the prominent beta and photon-emitting radionuclides (DIRS 154659-BSC 2001, Volume 2, Section 4.1.4, pp. 4 to 11)] in the representative volume of groundwater (3,000 acre-feet per year). The resulting concentrations for each time step were converted to equivalent doses by scaling the appropriate dose conversion factor (4 millirem per 2,000 picocurie per liter for carbon-14; 4 millirem per 1 picocurie per liter for iodine-129; and 4 millirem per 900 picocurie for technetium 99). Calculating the sum of these three radionuclide doses for each time step produced a time history of whole-body or organ dose; the peak within 10,000 years was identified and is reported in Table I-17. This process is repeated for 95th-percentile whole-body or organ dose using model-predicted 95th-percentile annual activity releases of the prominent beta and photon-emitting radionuclides.

I.6 Waterborne Chemically Toxic Material Impacts

Several materials that are chemically toxic would be used in the construction of the repository. A screening analysis was used to determine which, if any, of these materials would have the potential to be transported to the accessible environment in quantities sufficient to be toxic to humans.

Chemicals included in the substance list for the Environmental Protection Agency's Integrated Risk Information System (DIRS 103705-EPA 1997, all; DIRS 148219, 148221, 148224, 148227, 148228, 148229, and 148233-EPA 1999, all) were evaluated to determine a concentration that would be found in drinking water in a well downgradient from the repository. The chemicals on the Integrated Risk Information System substance list that would be in the repository are barium, boron, cadmium, chromium, copper, lead, manganese, mercury, molybdenum, nickel, selenium, uranium, vanadium, and zinc.

I.6.1 SCREENING ANALYSIS

The results of the analysis of long-term performance for radionuclides detailed in Section I.5 show that, at most, three waste packages would be breached prior to 10,000 years (due to improper heat treatment) under the Proposed Action. The period of consideration for chemical toxic materials impacts was 10,000 years. Therefore, only toxic materials outside the waste package were judged to be of concern in this analysis. These are chromium, copper, manganese, molybdenum, nickel, and vanadium.

I.6.1.1 Maximum Source Concentrations of Chemically Toxic Materials in the Repository

Maximum source concentrations were calculated to provide the maximum possible concentration of that element in water entering the unsaturated zone. For materials that were not principally part of the Alloy-22 (copper and manganese), the maximum source concentration was taken to be the solubility of the material in repository water. The solubilities were obtained by modeling with the EQ3 computer code (DIRS 100836-Wolery 1992, all). The simulations were started with water from well J-13 near the Yucca Mountain site (DIRS 100814-Harrar et al. 1990, all). EQ3 calculates chemical equilibrium of a system so that, by making successive runs with gradually increasing aqueous concentrations of an element,

Table I-15. Peak mean gross alpha activity for analyzed inventories, scenarios, and temperature operating modes.[a,b]

Modeled inventory, scenario[c], and operating mode	10,000 years		100,000 years		1 million years	
	Without background	With background[d]	Without background	With background	Without background	With background
Proposed Action, nominal, higher-temperature	1.8×10^{-8}	0.40	0.017	0.42	17.7	18.1
Proposed Action, nominal, lower-temperature	3.3×10^{-8}	0.40	0.010	0.41	14.2	14.6
Inventory Module 1, nominal, higher-temperature	3.3×10^{-8}	0.40	0.023	0.42	27.7	28.1
Inventory Module 2, nominal, higher-temperature	2.2×10^{-10}	0.40	0.000042	0.40	0.039	0.44
Proposed Action, human intrusion event at 30,000 years, higher-temperature	NA[e]	NA	0.00018	0.40	0.00031	0.40

a. Adapted from DIRS 157307-BSC (2001, Enclosure 1).
b. These results are based on an annual water usage equal to 3.7 million cubic meters (exactly 3000 acre-feet) per year.
c. Mean gross alpha activity is not available for igneous activity scenarios
d. Background alpha activity concentration is 0.4 picocurie per liter.
e. NA = not applicable.

Table I-16. Peak mean total radium concentration (picocuries per liter) for analyzed inventories, scenarios, and temperature operating modes.[a,b]

Modeled inventory, scenario[c], and operating mode	10,000 years		100,000 years		1 million years	
	Without background	With background[d]	Without background	With background	Without background	With background
Proposed Action, nominal, higher-temperature	1.1×10^{-11}	1.0	2.4×10^{-5}	1.0	0.33	1.4
Proposed Action, nominal, lower-temperature	2.4×10^{-12}	1.0	2.7×10^{-5}	1.0	0.27	1.3
Inventory Module 1, nominal, higher-temperature	3.3×10^{-10}	1.0	4.0×10^{-5}	1.0	0.67	1.7
Inventory Module 2, nominal, higher-temperature	6.7×10^{-13}	1.0	6.8×10^{-9}	1.0	0.0016	1.0
Proposed Action, human intrusion event at 30,000 years, higher-temperature	NA[e]	NA	2.4×10^{-7}	1.0	3.8×10^{-7}	1.0

a. Adapted from DIRS 157307-BSC (2001, Enclosure 1).
b. These results are based on an annual water usage equal to 3.7 million cubic meters (exactly 3000 acre-feet) per year.
c. Total radium concentration is not available for igneous activity scenarios
d. Background radium activity concentration is 1.04 picocuries per liter.
e. NA = not applicable.

Table I-17. Peak mean annual whole body or organ dose (millirem)[a] for the sum of all beta- and photon-emitting radionuclides during 10,000 years after closure for analyzed inventories, scenarios, and temperature operating modes.[b]

Modeled inventory, scenario and operating mode	Total
Proposed Action, nominal, higher-temperature	2.3×10^{-5}
Proposed Action, nominal, lower-temperature	1.3×10^{-5}
Proposed Action, human intrusion event at 30,000 years, higher-temperature	NA[c]
Inventory Module 1, nominal, higher-temperature operating mode	2.8×10^{-5}

a. This represents a bounding limit (overestimate) of the maximum dose to any organ because different radionuclides would affect different organs preferentially.
b. These results are based on an annual water usage equal to 3.7 million cubic meters (exactly 3000 acre-feet) per year.
c. NA = not applicable.

eventually a result will show the saturation of a mineral in that element. That concentration at which the first mineral saturates is said to be the "solubility." The solubility of copper (from the electrical bus bars left in the tunnels) was obtained by increasing copper concentrations in successive runs of EQ3. At a concentration of 0.018 milligram per liter, copper began to precipitate as tenorite (CuO). This mineral was then in equilibrium with dissolved copper existing in approximately equal molar parts as $CuOH^+$, $Cu(CO_3)aq$, and Cu^{++}. A similar approach for manganese gave a solubility of 4.4×10^{-10} milligram per liter as pyrolusite (MnO_2) began to precipitate. In the cases of chromium, molybdenum, nickel, and vanadium, the source concentration has a potential to be very high because the corrosion of Alloy-22 could result in a very low pH solution (much different from the repository water). Thus, for purposes of screening, it was assumed that these materials had a potentially very high source concentration and should be subjected to further screening analysis (this is discussed in Section I.6.2).

I.6.1.2 Further Screening for Chemically Toxic Materials

Manganese was further analyzed using a comparison of intake to the Oral Reference Dose. The Oral Reference Dose is an indication of the limit for possible health effects from oral ingestion. Intake was based on a 2-liter (0.53-gallon) daily consumption rate of drinking water, at the maximum source concentrations (solubilities), by a 70-kilogram (154-pound) adult. Calculation takes no credit for any dilution from the source to the recipient. For manganese, the intake would be 2.2×10^{-12} milligram per kilogram per day. This is very small compared to the Oral Reference Dose of 0.14 milligram per kilogram per day listed for manganese in the Integrated Risk Information System (DIRS 148227-EPA 1997, all). Thus, it is concluded that manganese requires no further consideration.

No Oral Reference Dose is available for copper, but a similar evaluation can be made by comparing the maximum source concentration to a maximum concentration limit for the drinking water standard (40 CFR 141.2). For copper the maximum contaminant limit is 1.3 milligrams per liter. This is much higher than the source concentration of 0.018 milligram per liter, so it is concluded that copper requires no further consideration.

ORAL REFERENCE DOSE

The *Oral Reference Dose* is based on the assumption that thresholds exist for certain toxic effects such as cellular necrosis. This dose is expressed in units of milligrams per kilogram per day. In general, the oral reference dose is an estimate (with uncertainty spanning perhaps an order of magnitude) of a daily exposure to the human population (including sensitive subgroups) that is likely to be without an appreciable risk of deleterious effects during a lifetime (DIRS 148219-EPA 1999, all).

The remaining hazardous elements of concern (chromium, molybdenum, nickel, and vanadium) are analyzed in the next section.

I.6.2 BOUNDING CONSEQUENCE ANALYSIS FOR CHEMICALLY TOXIC MATERIALS

Further evaluation is warranted because the first level of the screening analysis (Section I.6.1) indicated that the repository could release certain waterborne chemically toxic materials into groundwater in substantial quantities and that these could represent a potential human health impact. The following materials require further evaluation: chromium, molybdenum, nickel, and vanadium. A bounding calculation for concentrations in the biosphere is presented in this section for these elements that shows the impacts would be low enough to preclude any need for more detailed fate and transport analyses.

I.6.2.1 Assumptions

The following assumptions were applied to the bounding impact analysis for waterborne chemically toxic materials:

1. The general corrosion rate of Alloy-22 is equivalent for humid-air and aqueous corrosion conditions (this assumption is consistent with treatment of this substance in the TSPA–Site Recommendation).
2. The general corrosion rate of Alloy 316NG (stainless steel) is also equivalent for humid air and aqueous corrosion conditions.
3. Consistent with Assumptions 1 and 2 above, drip shields were not assumed to effectively delay onset of general corrosion of Alloy-22 in the outer barrier layer of waste packages or the emplacement pallets.
4. Consistent with Assumptions 1, 2, and 3 above, exposed Alloy-22 and stainless steel 316NG in the drip shield rail, waste packages, and emplacement pallets would all be subject to corrosion at the same time.
5. Consistent with Assumptions 1, 2, and 3 above, all waste packages would be subject to general corrosion at the same time, and would not experience variability in the time corrosion begins.
6. The median corrosion rates for Alloy-22 and stainless steel 316NG were used in the impact estimate calculations because the rates apply to all waste packages, drip shields, and emplacement pallets in the repository.
7. A migration pathway for mobilized waterborne chemically toxic materials through the engineered barrier system to the vadose zone was assumed to exist at all times general corrosion is in progress.
8. Time delays, mitigation effects by sorption in rocks, and other beneficial effects of transport in the geosphere were neglected for purposes of this bounding impact estimate; the mass of waterborne chemically toxic materials mobilized was assumed to be instantly available at the biosphere exposure locations.
9. The concentration in groundwater was estimated by diluting the released mass of waterborne chemically toxic materials in the representative volume defined by the Environmental Protection Agency [3.7 million cubic meters (exactly 3,000 acre-feet) of water per year] in 40 CFR Part 197.
10. Under the chemical environment of the waste package, all chromium, molybdenum, nickel, and vanadium were assumed to be in their most soluble and toxic state. This is a highly conservative assumption but is consistent with other modeling of the waste package chemical environment.
11. Mobilization of chromium, molybdenum, nickel, and vanadium was assumed equivalent to the corrosion loss of stainless steel or Alloy-22 times the fraction of each element in the alloys.
12. Throughout the discussions in Section I.6.2 it is assumed that the form of mobilized chromium is the hexavalent form. The hexavalent form of chromium [Cr(VI)] is considered potentially hazardous, whereas the more common corrosion product, trivalent chromium [Cr(III)], is not. This is a conservative assumption because DOE believes that most of the mobilized Cr would be the trivalent form.

I.6.2.2 Surface Area Exposed to General Corrosion

Corrosion of the materials bearing chromium and molybdenum would occur over all exposed surface areas. The total exposed surface area of Alloy-22 surfaces (drip shield rails, outer layer of waste packages, and portions of the emplacement pallets) and stainless-steel 316NG surfaces (portions of the emplacement pallets) are calculated in this section.

Tables I-18 and I-19 summarize the calculation of the total exposed surface areas for Alloy-22 contained in the waste packages and drip shields, respectively, under the Proposed Action.

Table I-18. Total exposed surface area of the Alloy-22 outer layer of all waste packages under the Proposed Action inventory.

Waste package type[a]	Number[b]	Outer diameter[a] (millimeters)[b]	Length[a] (millimeters)[b]	Surface area[c] (square millimeters)[d]	Total surface area (square meters)[e]
21 PWR absorber plate	4,299	1,664	5,165	31,349,978	134,774
21 PWR control rods	95	1,664	5,165	31,349,978	2,978
12 PWR absorber plate	163	1,330	5,651	26,390,258	4,302
44 BWR absorber plate	2,831	1,674	5,165	31,564,675	89,360
24 BWR thick absorber plate	84	1,318	5,105	23,866,529	2,005
5 DHLW/DOE SNF	1,592	2,110	3,590	30,790,593	49,019
5 DHLW/DOE SNF–long	1,751	2,110	5,217	41,575,586	72,799
Navy SNF	200	1,949	5,430	39,214,523	7,843
Navy SNF–long	100	1,949	6,065	43,102,606	4,310
2 MCO/2 HLW	186	1,815	5,217	34,921,842	6,495
Totals	***11,301***				***373,884***

a. Waste package data from DIRS 150558-CRWMS M&O (2000, all).
b. To convert millimeters to inches, multiply by 0.0394.
c. Surface area calculated as area of a right circular cylinder.
d. To convert square millimeters to square inches, multiply by 0.00155.
e. To convert square meters to square feet, multiply by 10.764.

Table I-19. Total exposed surface area of the Alloy-22 rails for all drip shields under the Proposed Action inventory.

Component	Number of pieces	Average waste package emplacement length[a] (millimeters)[b]	Width[c] (millimeters)[d]	Thickness[c] (millimeters)	Total surface area per average waste package[e] (square millimeters[f])	Number of waste packages[c]	Total surface area for repository (square meters[g])
Rail	2	5,076	115	10	1,370,520	11,301	15,488

a. Emplacement length estimate from DIRS 155393-CRWMS M&O (2000, Attachment V, p. V-2).
b. To convert meters to feet, multiply by 3.2808.
c. Waste package data from DIRS 150558-CRWMS M&O (2000, all).
d. To convert millimeters to inches, multiply by 0.0394.
e. Surface area calculated as sum of areas of wetted surfaces (two rectangles) of angles running along the bottom of both sides of the drip shield.
f. To convert square millimeters to square inches, multiply by 0.00155.
g. To convert square meters to square feet, multiply by 10.764.

Tables I-20 and I-21 summarize the calculation of the total exposed surface areas for Alloy-22 contained in the waste packages and drip shields, respectively, for the Module 1 inventory.

Tables I-22 and I-23 summarize the calculation of the total exposed surface areas for Alloy-22 contained in the waste packages and drip shields respectively, for the Module 2 inventory.

Table I-24 summarizes the calculation of total exposed surface area for the Alloy-22 components of the emplacement pallets for the Proposed Action, Module 1, and Module 2 inventories.

Table I-20. Total exposed surface area of the Alloy-22 outer layer of all waste packages for the Module 1 inventory.

Waste package type	Number[a]	Outer diameter[a] (millimeters)[b]	Length[a] (millimeters)	Surface area[c] (square millimeters)[d]	Total surface area (square meters)[e]
21 PWR absorber plate	6,733	1,664	5,165	31,349,978	211,079
21 PWR control rods	114	1,664	5,165	31,349,978	3,574
12 PWR absorber plate	390	1,330	5,651	26,390,258	10,292
44 BWR absorber plate	4,408	1,674	5,165	31,564,675	139,137
24 BWR thick absorber plate	109	1,318	5,105	23,866,529	2,601
5 DHLW/DOE SNF	1,557	2,110	3,590	30,790,593	47,941
5 DHLW/DOE SNF–long	2,821	2,110	5,217	41,575,586	117,285
Navy SNF	200	1,949	5,430	39,214,523	7,843
Navy SNF–long	100	1,949	6,065	43,102,606	4,310
2 MCO/2 HLW	199	1,815	5,217	34,921,842	6,949
Totals	***16,631***				***551,012***

a. Waste package data from DIRS 150558-CRWMS M&O (2000, all).
b. To convert millimeters to inches, multiply by 0.0394.
c. Surface area calculated as area of a right circular cylinder.
d. To convert square millimeters to square inches, multiply by 0.00155.
e. To convert square meters to square feet, multiply by 10.764.

Table I-21. Total exposed surface area of the Alloy-22 rails for all drip shields for the Module 1 inventory.

Component	Number of pieces	Average waste package emplacement length[a] (millimeters)[b]	Width[c] (millimeters)[d]	Thickness[c] (millimeters)[d]	Total surface area per average waste package[e] (square millimeters)[f]	Number of waste packages[c]	Total surface area for repository (square meters)[g]
Rail	2	5,076	115	10	1,370,520	16.631	22,793

a. Emplacement length estimate from DIRS 155393-CRWMS M&O (2000, Attachment V, p. V-2).
b. To convert meters to feet, multiply by 3.2808.
c. Waste package data from DIRS 150558-CRWMS M&O (2000, all).
d. To convert millimeters to inches, multiply by 0.0394.
e. Surface area calculated as sum of areas of wetted surfaces (two rectangles) of angles running along the bottom of both sides of the drip shield.
f. To convert square millimeters to square inches, multiply by 0.00155.
g. To convert square meters to square feet, multiply by 10.764.

Table I-22. Total exposed surface area of the Alloy-22 outer layer of all waste packages for the Module 2 inventory.

Waste package type	Number[a]	Outer diameter[a] (millimeters[b])	Length[a] (millimeters[b])	Surface area[c] (square millimeters[d])	Total surface area (square meters[e])
21 PWR absorber plate	6,733	1,664	5,165	31,349,978	211,079
21 PWR control rods	114	1,664	5,165	31,349,978	3,574
12 PWR absorber plate	390	1,330	5,651	26,390,258	10,292
44 BWR absorber plate	4,408	1,674	5,165	31,564,675	139,137
24 BWR thick absorber plate	109	1,318	5,105	23,866,529	2,601
5 DHLW/DOE SNF	1,557	2,110	3,590	30,790,593	47,941
5 DHLW/DOE SNF–long	2,821	2,110	5,217	41,575,586	117,285
Navy SNF	200	1,949	5,430	39,214,523	7,843
Navy SNF–long	100	1,949	6,065	43,102,606	4,310
Navy–long (GTCC and SPAR)[f]	601	1,949	6,065	43,102,606	25,905
2 MCO/2 DHLW	199	1,815	5,217	34,921,842	6,949
Totals	***17,232***				***576,917***

a. Waste package data from DIRS 150558-CRWMS M&O (2000, all).
b. To convert millimeters to inches, multiply by 0.0394.
c. Surface area calculated as area of a right circular cylinder.
d. To convert square millimeters to square inches, multiply by 0.00155.
e. To convert square meters to square feet, multiply by 10.764.
f. Navy SNF-long type waste packages used to represent disposal of Greater-Than-Class-C (GTCC) and Special-Performance-Assessment-Required (SPAR) waste.

Table I-23. Total exposed surface area of the Alloy-22 rails for all drip shields for the Module 2 inventory.

Component	Number of pieces	Average waste package emplacement length[a] (millimeters)[b]	Width[c] (millimeters)[d]	Thickness[c] (millimeters)[d]	Total surface area per average waste package[e] (square millimeters)[f]	Number of waste packages[c]	Total surface area for repository (square meters)[g]
Rail	2	5,076	115	10	1,370,520	17,232	23,617

a. Emplacement length estimate from DIRS 155393-CRWMS M&O (2000, Attachment V, p. V-2).
b. To convert meters to feet, multiply by 3.2808.
c. Waste package data from DIRS 150558-CRWMS M&O (2000, all).
d. To convert millimeters to inches, multiply by 0.0394.
e. Surface area calculated as sum of areas of wetted surfaces (two rectangles) of angles running along the bottom of both sides of the drip shield.
f. To convert square millimeters to square inches, multiply by 0.00155.
g. To convert square meters to square feet, multiply by 10.764.

Table I-24. Total exposed surface area of the Alloy-22 components for all emplacement pallets under the Proposed Action, Module 1, and Module 2 inventories.

Emplacement pallet component[a]	Number of pieces[a]	Length[a] (millimeters)[b]	Width[a] (millimeters)	Number of sides[a]	Total surface area per pallet (square meters)[c]	Number of pallets[d]	Total surface area repository (square meters)[c]
Plate 1	2	1,845	552.4	1	4.077[e]		
Plate 2	2	922.5	614	2	2.266[f]		
Plate 3	2				2.219[g]		
Plate 4	4	552	462	2	2.040[h]		
Plate 5	4	552	80	2	0.353[i]		
Plate 6	4	1,266.7	603.2	2	6.113[j]		
Plate 7	4	152.4	79.9	2	0.049[k]		
Plate 8	4	152.4	552.4	1	0.337[l]		
Totals for Proposed Action					***17.45***	***11,301***	***197,240***
Totals for Inventory Module 1					***17.45***	***16,631***	***290,266***
Totals for Inventory Module 2					***17.45***	***17,232***	***300,756***

a. Emplacement pallet details from sketches SK-0189 Rev 0 and SK-0144 Rev 1, DIRS 150558-CRWMS M&O (2000).
b. To convert millimeters to inches, multiply by 0.0394.
c. To convert square meters to square feet, multiply by 10.764.
d. Waste package data from DIRS 150558-CRWMS M&O (2000, all).
e. Calculated for one wetted rectangular side.
f. Calculated for both wetted rectangular sides.
g. Surface area equal to that of Plate 2 less area covered by 5.1-centimeter (2.0-inch) tube cross-sections.
h. Calculated assuming rectangular area covered by tubes is not wetted; note that while the inside and outside are covered by tubes the width dimension is correct for each side.
i. Calculated assuming rectangular wetted area.
j. Calculated assuming wetted area includes exposed edge thicknesses which are added to the length and width
k. Calculated based on triangular area.
l. Calculated assuming one wetted side only (because it is covered by the tube).

The sum of exposed total surface areas for waste packages, drip shield rails, and emplacement pallet components fabricated from Alloy-22 (from Tables I-18, I-19, and I-24) is 586,612 square meters (6,314,240 square feet) under the Proposed Action. For Inventory Module 1, the sum of exposed total surface areas (from Tables I-20, I-21, and I-24) is 864,072 square meters (9,300,794 square feet). For Inventory Module 2, the sum of exposed total surface areas (from Tables I-22, I-23, and I-24) is 901,290 square meters (9,701,400 square feet). This is the area of Alloy-22 subject to generalized corrosion under the assumptions outlined for this bounding impact estimate.

Tables I-25, I-26, and I-27 summarize the calculation of the total exposed surface areas for stainless steel 316NG used in the emplacement pallets for the Proposed Action, Module 1, and Module 2 inventories, respectively.

Table I-25. Total exposed surface area of the stainless-steel 316NG components for all emplacement pallets under the Proposed Action inventory.

Emplacement pallet tubes	Number of pieces[a]	Length[a] (millimeters[b])	Width[a] (millimeters)	Number of sides[a]	Total surface area per average waste package[c] (square meters[d])	Number of waste packages[e,f]	Total surface area repository (square meters)
Long pallets	4	4,147	609.6	2	18.877[f]	9,709	183,278
Short pallets	4	2,500	609.6	2	10.845[g]	1,592	17,265
Totals						***11,301***	***200,543***

a. Emplacement pallet details from sketches SK-0189 Rev 0 and SK-0144 Rev 1 (DIRS 150558-CRWMS M&O 2000).
b. To convert millimeters to inches, multiply by 0.0394.
c. Calculated for area of all wetted rectangular sides.
d. To convert square meters to square feet, multiply by 10.764.
e. Waste package data from DIRS 150558-CRWMS M&O (2000, all).
f. Only waste packages of type "5 DHLW/DOE SNF" are assumed to utilize short pallets.

Table I-26. Total exposed surface area of the stainless-steel 316NG components for all emplacement pallets for the Module 1 inventory.

Emplacement pallet tubes	Number of pieces[a]	Length[a] (millimeters[b])	Width[a] (millimeters)	Number of sides[a]	Total surface area per average waste package[c] (square meters[d])	Number of waste packages[e,f]	Total surface area repository (square meters)
Long pallets	4	4,147	609.6	2	18.877[f]	15,075	284,533
Short pallets	4	2,500	609.6	2	10.845[g]	1,557	16,886
Totals						***16,632***	***301,419***

a. Emplacement pallet details from sketches SK-0189 Rev 0 and SK-0144 Rev 1 (DIRS 150558-CRWMS M&O 2000).
b. To convert millimeters to inches, multiply by 0.0394.
c. Calculated for area of all wetted rectangular sides.
d. To convert square meters to square feet, multiply by 10.764.
e. Waste package data from DIRS 150558-CRWMS M&O (2000, all).
f. Only waste packages of type "5 DHLW/DOE SNF" are assumed to utilize short pallets.

Table I-27. Total exposed surface area of the stainless-steel 316NG components for all emplacement pallets for the Module 2 inventory.

Emplacement pallet tubes	Number of pieces[a]	Length[a] (millimeters[b])	Width[a] (millimeters)	Number of sides[a]	Total surface area per average waste package[c] (square meters[d])	Number of waste packages[e,f]	Total surface area repository (square meters)
Long pallets	4	4,147	609.6	2	18.877[f]	15,675	295,899
Short pallets	4	2,500	609.6	2	10.845[g]	1,557	16,886
Totals						***17,232***	***312,785***

a. Emplacement pallet details from sketches SK-0189 Rev 0 and SK-0144 Rev 1 (DIRS 150558-CRWMS M&O 2000).
b. To convert millimeters to inches, multiply by 0.0394.
c. Calculated for area of all wetted rectangular sides.
d. To convert square meters to square feet, multiply by 10.764.
e. Waste package data from DIRS 150558-CRWMS M&O (2000, all).
f. Only waste packages of type "5 DHLW/DOE SNF" are assumed to utilize short pallets.

I.6.2.3 General Corrosion Rates

The general corrosion rate for Alloy-22 has been measured in laboratory experiments. The corrosion rate was input to the TSPA model as a cumulative distribution function. The 5th percentile is 0.000012 millimeter (0.0000004 inch) per year, the median value is 0.000045 millimeter (0.0000017 inch) per year, and the 95th-percentile of the distribution is 0.00008 millimeter (0.000003 inch) per year (DIRS 152542-CRWMS M&O 2000, Figure 1, p. 11). For purposes of this bounding calculation, the median rate was chosen because the calculation is concerned with the average rate of corrosion over a large number of waste packages, drip shield rails, and emplacement pallets. Hence, the median rate is representative of repository conditions taken as a whole over the 10,000-year post-closure period.

The median general corrosion rate for stainless steel 316NG is 0.01 micron per year (0.0000394 inch per year) (DIRS 135968-CRWMS M&O 2000, Figure 3-15, p. 3-30).

I.6.2.4 Release Rates

The rate of release of waterborne chemically toxic materials was calculated as the product of the surface area exposed to general corrosion, the general corrosion rate, and the weight fraction of the alloy for the waterborne chemically toxic material of interest. Alloy-22 is comprised of among other elements, 22.5 percent (maximum) chromium, 14.5 percent (maximum) molybdenum, 57.2 percent nickel, and 0.35 percent vanadium (DIRS 104328-ASTM 1998, all). Stainless steel 316NG is assumed to be essentially the same as 316L, which is comprised, among other elements, of 17.0 percent chromium, 12 percent nickel, and 2.5 percent molybdenum with no vanadium (DIRS 102933-CRWMS M&O 1999, p. 13).

Tables I-28, I-29, and I-30 summarize the calculation of the bounding mass release rates for the Proposed Action, Module 1, and Module 2 inventories, respectively. The mass release rates for chromium, molybdenum, nickel, and vanadium are based on the surface exposure area of exposed repository components containing these elements, the general corrosion rates for those components, and the weight percent content of the individual elements.

Table I-28. Bounding mass release rates (grams per year)[a] from Alloy-22 and stainless-steel 316NG components from general corrosion for the Proposed Action.

Alloy	Total exposed surface area in repository (square meters)[b]	General corrosion rate (meters per year)[c]	Alloy release volume (cubic meter per year)[d]	Alloy density (grams per cubic meter)[e]	Bounding mass release rate (grams per year)[a]				
					Alloy	Chromium	Molybdenum	Nickel	Vanadium
Alloy-22	586,612	4.5×10^{-8}	0.0264	8,690,000	229,395	51,614	33,262	131,099	803
316NG	200,543	1.0×10^{-8}	0.00201	7,980,000	16,003	2,721	400	1,920	0
Totals						***54,334***	***33,662***	***133,019***	***803***

a. To convert grams to ounces, multiply by 0.035273.
b. To convert square meters to square feet, multiply by 10.764.
c. To convert meters to feet, multiply by 3.2468.
d. To convert cubic meters to cubic feet, multiply by 35.314.
e. To convert grams per cubic meter to ounces per cubic foot, multiply by 0.0010047.

Table I-29. Bounding mass release rates (grams per year)[a] from Alloy-22 and stainless-steel 316NG components from general corrosion for Module 1.

Alloy	Total exposed surface area in repository (square meters)[b]	General corrosion rate (meters per year)[c]	Alloy release volume (cubic meter per year)[d]	Alloy density (grams per cubic meter)[e]	Bounding mass release rate (grams per year)[a]				
					Alloy	Chromium	Molybdenum	Nickel	Vanadium
Alloy-22	864,072	4.5×10^{-8}	0.0389	8,690,000	337,895	76,026	48,995	193,107	1,183
316NG	312,785	1.0×10^{-8}	0.0030	7,980,000	24,055	4,089	601	2,887	0
Totals						***80,116***	***49,596***	***195,994***	***1,183***

a. To convert grams to ounces, multiply by 0.035273.
b. To convert square meters to square feet, multiply by 10.764.
c. To convert meters to feet, multiply by 3.2468.
d. To convert cubic meters to cubic feet, multiply by 35.314.
e. To convert grams per cubic meter to ounces per cubic foot, multiply by 0.0010047.

I.6.2.5 Summary of Bounding Impacts

The bounding maximum concentration is based on the general corrosion rate of the source materials and the representative volume for dilution prescribed in the final Environmental Protection Agency regulation 40 CFR Part 197. Diluting the bounding release rates presented in Section I.6.2.4 for chromium, molybdenum, nickel, and vanadium in the prescribed representative volume of water (3.7 million cubic meters, or exactly 3,000 acre-feet per year) used for calculation of groundwater protection impacts for

Table I-30. Bounding mass release rates (grams per year)[a] from Alloy-22 and stainless-steel 316NG components from general corrosion for Module 2.

Alloy	Total exposed surface area in reposritoy (square meters)[b]	General corrosion rate (meters per year)[c]	Alloy release volume (cubic meter per year)[d]	Alloy density (grams per cubic meter)[e]	Bounding mass release rate (grams per year)[a]				
					Alloy	Chromium	Molybdenum	Nickel	Vanadium
Alloy-22	901,290	4.5×10^{-8}	0.0406	8,690,000	352,450	79,301	51,105	201,425	1,233
316NG	312,785	1.0×10^{-8}	0.0031	7,980,000	24,960	4,243	624	2,995	0
Totals						***83,544***	***51,729***	***204,420***	***1,233***

a. To convert grams to ounces, multiply by 0.035273.
b. To convert square meters to square feet, multiply by 10.764.
c. To convert meters to feet, multiply by 3.2468.
d. To convert cubic meters to cubic feet, multiply by 35.314.
e. To convert grams per cubic meter to ounces per cubic foot, multiply by 0.0010047.

waterborne radioactive materials results in the bounding concentration in groundwater at exposure locations for these chemically toxic materials listed in Table I-31.

Table I-31. Bounding concentrations of waterborne chemical materials of concern compared to Maximum Contaminant Levels Goals (milligrams per liter).

Material	Maximum Contaminant Level Goal	Maximum bounding concentration		
		Proposed Action	Inventory Module 1	Inventory Module 2
Chromium (VI)	0.1[a]	0.015	0.022	0.023
Molybdenum	NA[b]	0.009	0.013	0.014
Nickel	NA	0.036	0.053	0.055
Vanadium	NA	0.00022	0.00032	0.00033

a. 40 CFR 141.51.
b. NA = not available.

There are two measures for comparing human health effects for chromium. When the Environmental Protection Agency established its Maximum Contaminant Level Goals, it considered safe levels of contaminants in drinking water and the ability to achieve these levels with the best available technology. The Maximum Contaminant Level Goal for chromium is 0.1 milligram per liter (40 CFR 141.51). The bounding concentrations for the Proposed Action and for Inventory Modules 1 and 2 (Table I-31) are well below the Maximum Contaminant Level Goal for chromium. The other measure for comparison is the Oral Reference Dose for chromium, which is 0.005 milligram per kilogram of body mass per day (DIRS 148224-EPA 1999, all). The reference dose factor represents a level of intake that has no adverse effect on humans. It can be converted to a threshold concentration level for drinking water. The conversion yields essentially the same concentration for the reference dose factor as the Maximum Contaminant Level Goal.

No attempt can be made at present to express the bounding estimate of groundwater concentration of hexavalent chromium in terms of human health effects (for example, latent cancer fatalities). The carcinogenicity of hexavalent chromium by the oral route of exposure cannot be determined because of a lack of sufficient epidemiological or toxicological data (DIRS 148224-EPA 1999, all; DIRS 101825-EPA 1998, p. 48).

There is no Maximum Contaminant Level Goals for molybdenum, nickel, or vanadium. However, we can compare the intake based on the maximum bounding concentrations in Table I-31 to the Oral Reference Dose for each of these materials. The intakes by chemical, assuming water consumption of 2 liters (0.53 gallon) per day by a 70-kilogram (154-pound) person, are listed in Table I-32 along with the relevant Oral Reference Dose. The values in Table I-32 show that the intakes are well below the respective Oral Reference Doses for chromium, molybdenum, nickel, and vanadium for the Proposed Action, Inventory Modules 1, and Inventory Module 2.

Table I-32. Summary of intake of waterborne chemical materials of concern based on maximum bounding concentrations listed in Table I-31 compared to Oral Reference Doses.

Material	Oral Reference Dose	Intake[a]		
		Proposed Action	Inventory Module 1	Inventory Module 2
Chromium (VI)	0.005[b]	0.00042	0.00062	0.00065
Molybdenum	0.005[c]	0.00026	0.00038	0.00040
Nickel	0.02[d]	0.0010	0.0015	0.0016
Vanadium	0.007[e]	0.0000062	0.0000091	0.000010

a. Assuming daily intake of 2.0 liters (0.53 gallon) per day by a 70-kilogram (154-pound) individual.
b. DIRS 148224-EPA 1999, all.
c. DIRS 148228-EPA 1999, all.
d. DIRS 148229-EPA 1999, all.
e. DIRS 103705-EPA 1997, all.

Because the bounding concentration of chromium, molybdenum, nickel, and vanadium in groundwater is calculated to be below the Maximum Contaminant Level Goal or yield intakes well below the respective Oral Reference Doses, there is no further need to refine the calculation to account for physical processes that would limit mobilization of these materials or delay and dilute them during transport in the geosphere.

I.7 Atmospheric Radioactive Material Impacts

Following closure of the proposed Yucca Mountain Repository, there would be limited potential for releases to the atmosphere because the waste would be isolated far below the ground surface. Still, the rock is porous and does allow gas to flow, so the analysis must consider possible airborne releases. The only radionuclide that would have a relatively large inventory and a potential for gas transport is carbon-14. Iodine-129 can exist in a gas phase, but it is highly soluble and, therefore, would be more likely to dissolve in groundwater rather than migrate as a gas. Other gas-phase isotopes were eliminated in the screening analysis (Section I.3.3), usually because they have short half-lives and are not decay products of long-lived isotopes. A separate screening argument for radon-222 is provided in Section I.7.3. After carbon-14 escaped from the waste package, it could flow through the rock in the form of carbon dioxide. Atmospheric pathway models were used to estimate human health impacts to the local population in the 80-kilometer (50-mile) region surrounding the repository.

About 2 percent of the carbon-14 in commercial spent nuclear fuel exists as a gas in the space (or *gap*) between the fuel and the cladding around the fuel (DIRS 103446-Oversby 1987, p. 92). The average carbon-14 inventory in a commercial spent nuclear fuel waste package is approximately 1.37 grams (0.048 ounce) (6.11 curies) (see Table I-5), so the analysis used a gas-phase inventory of 0.122 curie of carbon-14 per commercial spent nuclear fuel waste package to calculate impacts from the atmospheric release pathway. The waterborne radioactive materials analysis described in Chapter 5, Section 5.4 included the entire inventory of the carbon-14 in the repository in the groundwater release models. Thus, the groundwater-based impacts would be overestimated slightly (by 2 percent) by this modeling approach.

Carbon is the second-most abundant element (by mass) in the human body, constituting 23 percent of Reference Man (DIRS 101074-ICRP 1975, p. 327). Ninety-nine percent of the carbon comes from food ingestion (DIRS 148066-Killough and Rohwer 1978, p. 141). Daily carbon intakes are approximately 300 grams (0.7 pound) and losses include 270 grams (0.6 pound) exhaled, 7 grams (0.02 pound) in feces, and 5 grams (0.01 pound) in urine (DIRS 101074-ICRP 1975, p. 377).

Carbon-14 dosimetry can be performed assuming specific-activity equivalence. The primary human intake pathway of carbon is food ingestion. The carbon-14 in food results from photosynthetic processing of atmospheric carbon dioxide, whether the food is the plant itself or an animal that feeds on the plant. Biotic systems, in general, do not differentiate between carbon isotopes. Therefore, the carbon-14 activity concentration in the atmosphere will be equivalent to the carbon-14 activity concentration in the plant, which in turn will result in an equivalent carbon-14 specific activity in human tissues.

I.7.1 CARBON-14 RELEASES TO THE ATMOSPHERE

The calculation of regional radiological doses requires estimation of the annual release rate of carbon-14. The analysis based the carbon-14 release rate on the estimated timeline of container failures for the higher-temperature repository operating mode, using the time-dependent mean value of the number of failed waste packages. The expected number of commercial spent nuclear fuel waste package failures in 100-year intervals was used to estimate the carbon-14 release rate after repository closure. The estimated amount of material released from each package as a function of time was reduced to account for radiological decay.

As for the waterborne radioactive material releases described in Chapter 5, Section 5.4, credit was taken for the intact zirconium-alloy cladding (on approximately 99 percent by volume of the commercial spent nuclear fuel at emplacement) delaying the release of gas-phase carbon-14. The remaining 1 percent by volume of the commercial spent nuclear fuel either would have stainless-steel cladding (which degrades much more quickly than zirconium alloy) or would already have failed in the reactor. The cladding failure submodel of the TSPA model also estimates the time of the first perforation through the cladding. Because carbon-14 in gas form as carbon dioxide can migrate through small holes, the time of first perforation was used as the time of release from the carbon-14 from the failed fuel element. A plot of the fraction of the cladding that has been perforated as a function of time after repository closure is shown in Figure I-27.

The amount (in curies) of carbon-14 that would be available for transport, A_T, from a waste package at the time it fails is calculated as:

$$A_T = D_F \times F_{FC} \times 0.122 \text{ curies per package}$$

where:

D_F = Time-dependent factor that accounts for radioactive decay (unitless)

F_{FC} = Fraction of perforated cladding (unitless)

The analysis technique calculated the above quantity on a time interval of every 100 years. At each time interval, the amount of carbon-14, B_T, available for transport due to further cladding perforations in waste packages that failed previously was also calculated. This amount was calculated as follows:

$$B_T = D_F \times DF_{FC} \times N_{PF} \times 0.122 \text{ curies per package}$$

where:

DF_{FC} = Fraction of cladding that was perforated in the 100-year time interval (unitless)

N_{PF} = Number of waste packages that had failed prior to the current 100-year time interval (unitless)

Rather than conducting a detailed gas-flow model of the mountain, the analysis assumed that the carbon-14 from the failed waste package would be released to the ground surface uniformly over a

100-year interval. Thus, the release rate (curies per year) to the ground surface, G_S, for a time interval was calculated as follows:

$$G_S = (N_{CI} \times A_T + B_T) / 100$$

where:

N_{CI} = Number of waste packages that failed in the current 100-year time interval (unitless)

Figure I-28 shows the estimated release rate of carbon-14 from the repository for 80,000 years after repository closure, assuming that the commercial spent nuclear fuel with perforated cladding had released its gas-phase carbon-14 prior to being placed in a waste package. The results in Figure I-28 are based on the Proposed Action inventory. Each symbol in the figure represents the carbon-14 release rate to the ground surface for a period of 100 years. The general downward slope of the symbols is due to radioactive decay (carbon-14 has a half-life of 5,730 years). The symbols indicating near-zero releases (curies per year) indicate that no waste packages failed during some 100-year periods, and the fraction of perforated cladding changed only a small amount. Using this expected-value representation of waste package lifetime, only 1 of 7,860 commercial spent nuclear fuel waste packages would have failed during the first 10,000 years after repository closure. See Section I.2.4 for a description of early waste package failure mechanisms. The second waste package would fail at about 53,000 years after repository closure. By 80,000 years after repository closure, 131 of the 7,860 commercial spent nuclear fuel waste packages would have failed. Using this expected-value representation of the time of first cladding perforation, about 2 percent of the cladding would be perforated in the first 10,000 years. Thus, all releases prior to 50,000 years on Figure I-28 come from a single waste package. The maximum release rate would occur about 1,700 years after repository closure. The estimated maximum release rate would be about 3.3 microcurie per year.

For Inventory Module 1, the number of idealized waste packages containing commercial spent nuclear fuel would increase from 7,860 to 11,754. Using the expected value curves for waste package failure, there would only be 1 waste package failure in the first 10,000 years for Inventory Module 1. Even though the modeled time of the waste package failure is 100 years earlier than for the Proposed Action inventory, the expected value for the fraction of cladding perforated is nearly identical for the two inventory modules during the first 10,000 years. Thus, the maximum release rate to the ground surface is the same and occurs at the same time for both inventory modules. Inventory Module 2 would not add any additional materials expected to contain gas-phase carbon-14, so it would have the same maximum release rate to the ground surface as the Proposed Action inventory.

I.7.2 ATMOSPHERE CONSEQUENCES TO THE LOCAL POPULATION

DOE used the GENII program (DIRS 100953-Napier et al. 1998, all) to model the atmospheric transport and human uptake of released carbon-14 for the 80-kilometer (50-mile) population radiological dose calculation. Radiological doses to the regional population near Yucca Mountain from carbon-14 releases were estimated using the population distribution described in Appendix G, Section G.2.1, which indicates approximately 76,000 people would live in the region surrounding Yucca Mountain in 2035. The population by distance and sector used in the calculations are listed in Table G-48. The computation also used current (1993 to 1996) annual average meteorology. The joint frequency data are listed in Table I-33.

A population radiological dose factor of 4.6×10^{-9} person-rem per microcurie per year of release was calculated using the GENII code. For a 3.3-microcurie-per-year maximum release rate, an 80-kilometer (50-mile) population radiological dose rate would be 1.5×10^{-8} person-rem per year. This radiological dose rate represents 7.5×10^{-12} latent cancer fatality in the regional population of 76,000 persons each

Table I-33. Meteorologic joint frequency data used for Yucca Mountain atmospheric releases (percent of time).[a]

Average wind speed (m/s)[b]	Atmospheric stability class	Direction (wind toward)															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
0.9	A	0.807	0.633	0.613	0.520	0.462	0.604	0.688	0.659	0.467	0.340	0.183	0.200	0.197	0.212	0.412	0.778
	B	0.279	0.479	0.392	0.325	0.372	0.540	1.243	2.279	1.484	0.499	0.290	0.192	0.105	0.070	0.087	0.305
	C	0.113	0.105	0.064	0.017	0.015	0.020	0.041	0.157	0.122	0.067	0.055	0.020	0.012	0.020	0.009	0.032
	D	0.003	0.003	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2.55	A	0.099	0.073	0.026	0.020	0.026	0.017	0.023	0.061	0.041	0.029	0.023	0.017	0.029	0.029	0.052	0.096
	B	0.058	0.044	0.038	0.026	0.032	0.061	0.125	0.377	0.360	0.070	0.049	0.015	0.009	0	0.009	0.017
	C	0.229	0.267	0.256	0.116	0.110	0.105	0.328	1.193	2.404	0.909	0.671	0.302	0.157	0.142	0.125	0.174
	D	0.105	0.049	0.038	0.003	0.003	0.003	0.006	0.035	0.444	0.290	0.206	0.055	0.035	0.049	0.087	0.099
	E	0.003	0.006	0	0.003	0	0	0.003	0.003	0.003	0.006	0.003	0.003	0.003	0.003	0	0.003
	F	0	0.003	0	0	0	0	0	0	0.003	0.003	0	0	0	0	0	0.003
4.35	A	0.096	0.096	0.041	0.015	0.012	0.009	0.015	0.023	0.058	0.044	0.026	0.023	0.029	0.020	0.020	0.070
	B	0.052	0.087	0.041	0.023	0.006	0.026	0.078	0.261	0.305	0.131	0.076	0.017	0.006	0.003	0.009	0.032
	C	0.142	0.241	0.168	0.070	0.029	0.076	0.131	0.740	1.638	0.308	0.290	0.119	0.049	0.041	0.038	0.102
	D	0.253	0.264	0.163	0.049	0.020	0.020	0.020	0.392	2.375	0.447	0.285	0.081	0.046	0.058	0.139	0.346
	E	0.006	0.017	0	0	0	0	0	0.003	0.006	0.020	0.015	0.006	0.003	0.003	0.012	0.020
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6.95	A	1.568	0.642	0.215	0.038	0.035	0.009	0.023	0.026	0.081	0.142	0.261	0.163	0.209	0.314	0.343	0.819
	B	0.682	0.552	0.067	0.003	0.006	0.006	0.023	0.058	0.348	0.325	0.267	0.131	0.078	0.093	0.078	0.256
	C	0.993	0.560	0.105	0.012	0.009	0.078	0.090	0.244	0.984	0.526	0.337	0.192	0.067	0.076	0.073	0.189
	D	1.594	0.912	0.183	0.020	0.020	0.006	0.035	0.566	3.368	0.430	0.160	0.128	0.035	0.044	0.142	0.598
	E	0.735	0.366	0.067	0.012	0.006	0	0	0.386	2.515	0.192	0.038	0.015	0	0.015	0.064	0.804
	F	0.238	0.096	0.003	0	0.003	0	0	0.142	1.641	0.055	0.032	0	0.003	0.003	0.029	0.796
9.75	A	2.134	0.935	0.218	0.078	0.029	0.041	0.026	0.070	0.163	0.232	0.203	0.232	0.267	0.372	0.587	1.388
	B	0.865	0.627	0.081	0.009	0.003	0.017	0.020	0.046	0.319	0.267	0.154	0.131	0.070	0.052	0.113	0.302
	C	0.720	0.261	0.038	0.012	0.020	0.020	0.009	0.076	0.502	0.299	0.148	0.229	0.078	0.032	0.041	0.157
	D	0.415	0.212	0.020	0.003	0.003	0.003	0.003	0.046	0.627	0.154	0.044	0.032	0.029	0.009	0.026	0.145
	E	0.029	0.006	0	0	0.003	0	0	0	0.006	0.003	0.003	0	0	0.003	0	0.003
	F	0	0.003	0	0	0	0	0	0	0	0	0	0	0	0.003	0	0.003
12.98	A	1.661	0.706	0.418	0.322	0.247	0.244	0.366	0.343	0.407	0.380	0.302	0.299	0.357	0.537	1.083	2.038
	B	0.836	0.668	0.253	0.107	0.157	0.116	0.264	0.499	0.674	0.404	0.270	0.171	0.122	0.096	0.232	0.950
	C	0.322	0.267	0.087	0.017	0.006	0.012	0.026	0.136	0.311	0.107	0.032	0.029	0.020	0.009	0.015	0.038
	D	0.006	0.006	0	0	0	0	0	0.003	0.012	0.003	0	0	0	0	0	0.003
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

a. Source: Adapted from data in DIRS 102877-CRWMS M&O (1999, Appendix B, all).

b. m/s = meters per second; to convert meters per second to miles per hour, multiply by 2.237.

year at the maximum release rate. This annual population radiological dose rate corresponds to a lifetime radiological population dose of 1.1×10^{-6} rem (assuming a 70-year lifetime), which corresponds to 5.3×10^{-10} latent cancer fatality during the 70-year period of the maximum release.

The impacts were also calculated for a maximally exposed individual. Given the population data in Appendix G, Table G-48 and the joint frequency data in Table I-33, the maximally exposed individual would reside 24 kilometers (15 miles) south of the repository. An individual radiological dose factor of 5.6×10^{-14} rem per microcurie per year of release was calculated using the GENII code for this location. For a 3.3-microcurie-per-year maximum release rate, the individual maximum radiological dose rate would be 1.8×10^{-13} rem per year, corresponding to a 9.2×10^{-17} probability of a latent cancer fatality. The 70-year lifetime dose would be 1.3×10^{-11} rem, representing a 6.4×10^{-15} probability of a latent cancer fatality.

I.7.3 SCREENING ARGUMENT FOR RADON

The uranium placed in the repository would continuously produce radon as a decay product. The longest-lived radon isotope is radon-222, with a half-life of 4 days (DIRS 103178-Lide and Frederikse 1997, p. 4-24). The only potential transport and human exposure pathway for radon would be through the atmosphere because radon would not travel far enough in water to reach an individual before decaying.

A study performed by Y.S. Wu and others (DIRS 103690-Wu, Chen, and Bodvarsson 1995, all) at Lawrence Berkeley National Laboratory calculated gas and heat flow from the mountain due to steam formation and repository induced heating. This study calculated heat and mass fluxes for 57- and 114-kilowatt-per-acre emplacements. The study indicated maximum gas fluxes at the surface of about 2×10^{-7} kilogram per second per square meter at the Ghost Dance and Solitario Canyon faults and generally no more that 2×10^{-9} kilogram per second per square meter over the remainder of the surface.

The gas flux at the Ghost Dance fault was used to estimate a lower limit for the gas travel time after the waste packages began to fail. The travel times would be longer for a smaller thermal gradient and most waste packages are estimated to remain intact until long after the thermal gradient from the waste emplacement had declined to almost zero. However, this calculation still applies if a waste package failed during the period of highest thermal gradient.

A gas pore velocity, using the estimated gas flux for the Ghost Dance Fault, applicable for gas travel from the repository horizon to the surface, is calculated from the following equation:

$$Vp = Fg / (Da \times Rp)$$

where:

Fg = Gas flux (2×10^{-7} kilogram per second per meter squared)

Da = Density of air (approximately 1.2 kilogram per cubic meter at 20º Celsius) (DIRS 127163-Weast 1972, p. F-11)

Rp = Rock porosity (0.082, unitless) (DIRS 100033-Flint 1998, Table 7, p. 44)

Vp = Pore Velocity (meters per second) = 2.03×10^{-6}

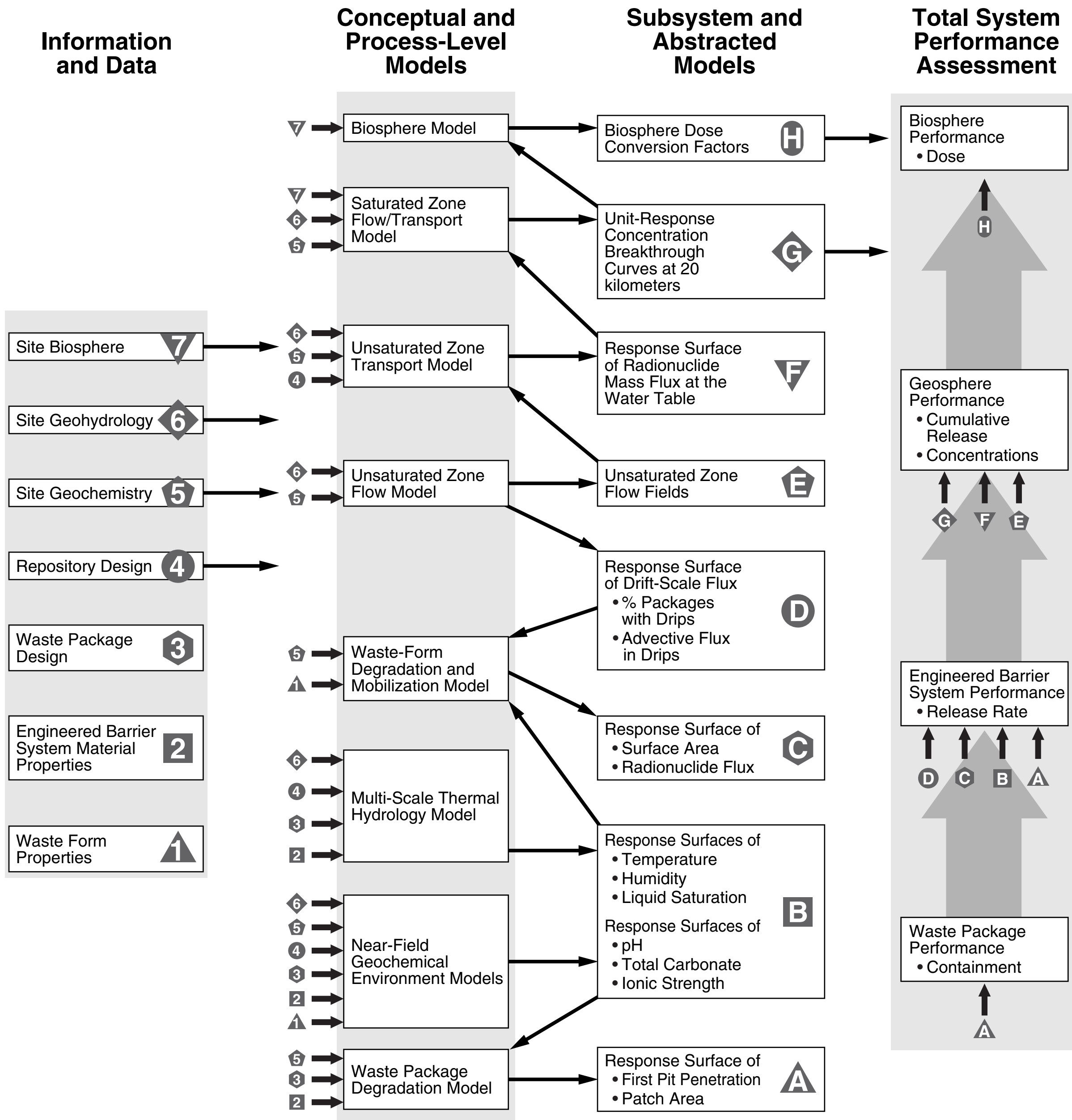
Travel time from the repository horizon to the surface is calculated from the following equation:

$$T_t = Rd / (Vp \times 86400)$$

where:

Rd	=	Depth to the repository (approximately 200 meters)
86400	=	Number of seconds per day
T_t	=	Gas travel time (days) = 1,140

Because the radioactive decay constant for radon-222 is 0.18145 (per day), radioactive decay would reduce the amount of radon-222 in the air by approximately 90 orders of magnitude in the time it took the air to travel from the repository horizon up through 200 meters (660 feet) of overlying rock. Therefore, no human effects are anticipated from the atmospheric release of radon-222 in the waste packages.



Source: Adapted from DIRS 153246-CRWMS M&O 2000, Figure 2.2-1, p. F2-18.

Figure I-1. TSPA model.

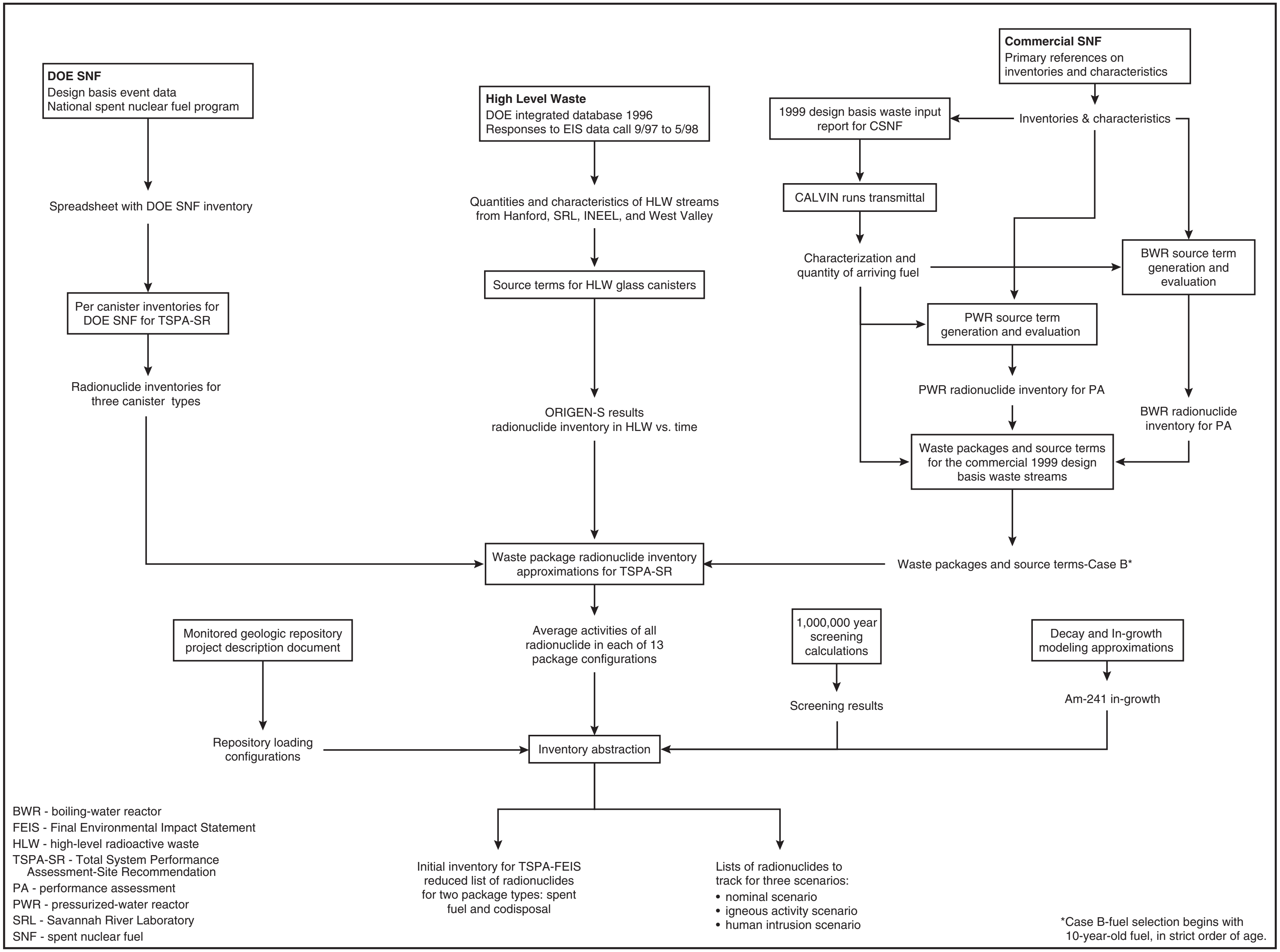


Figure I-2. Development of abstracted inventory for TSPA-FEIS.

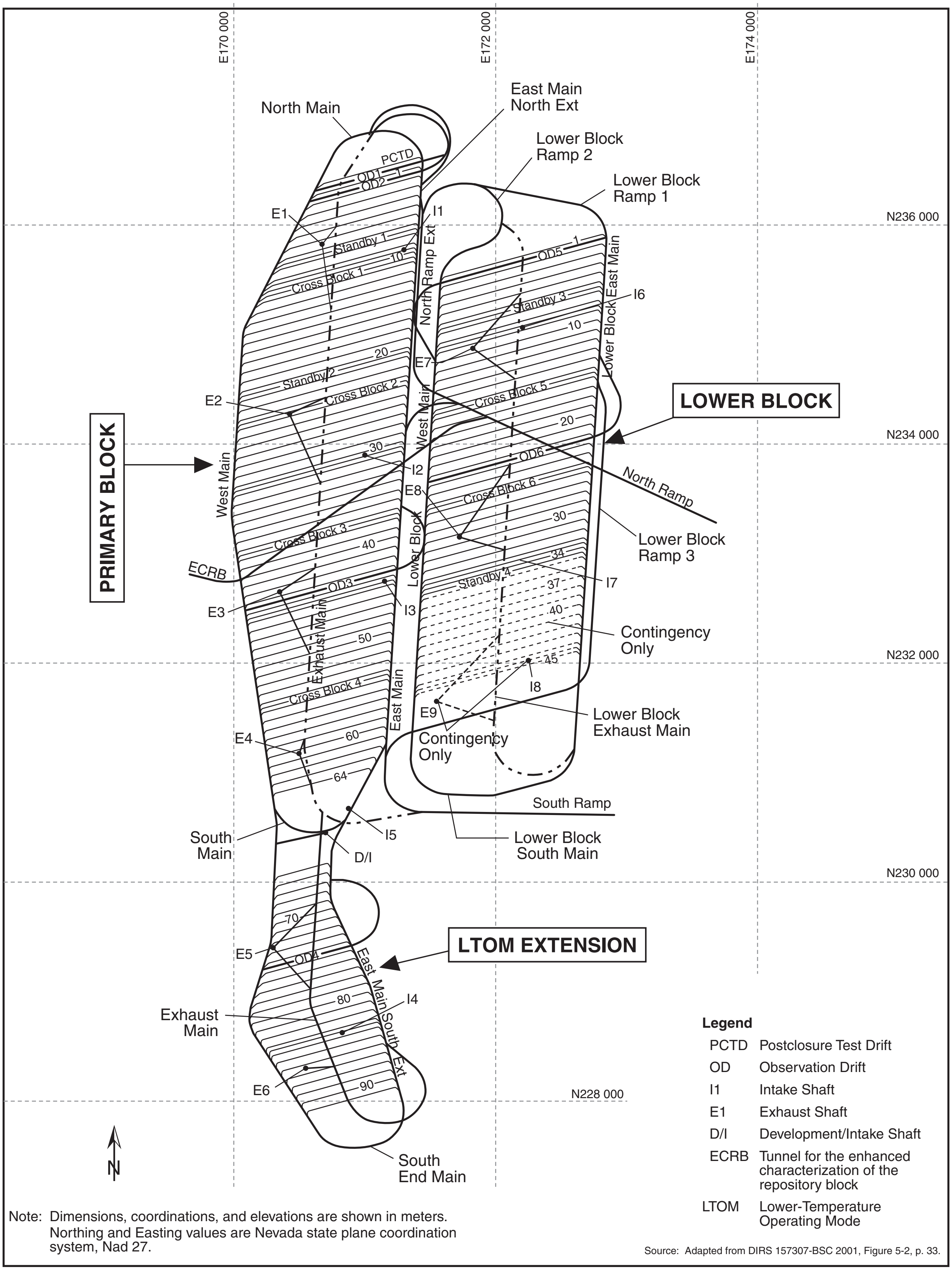
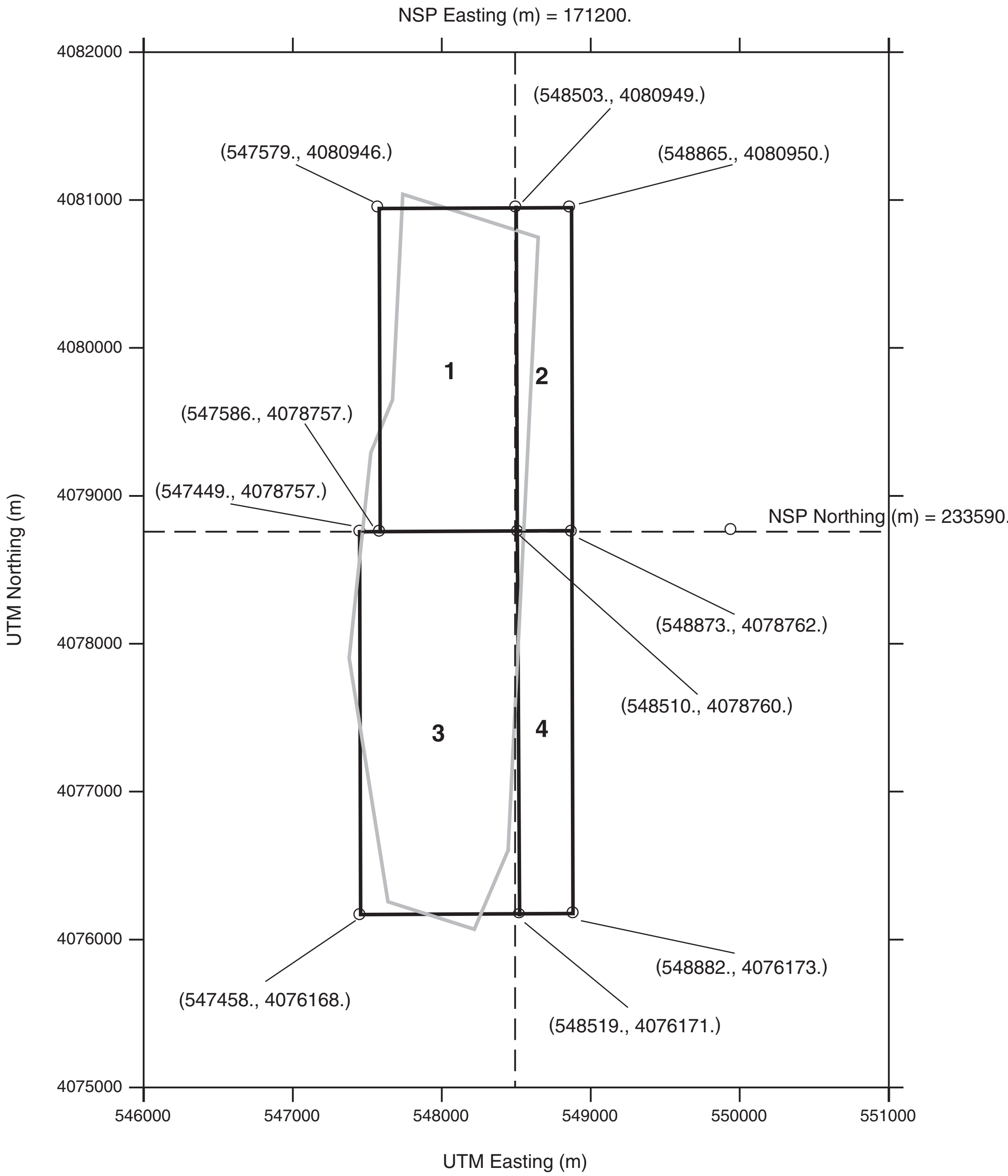


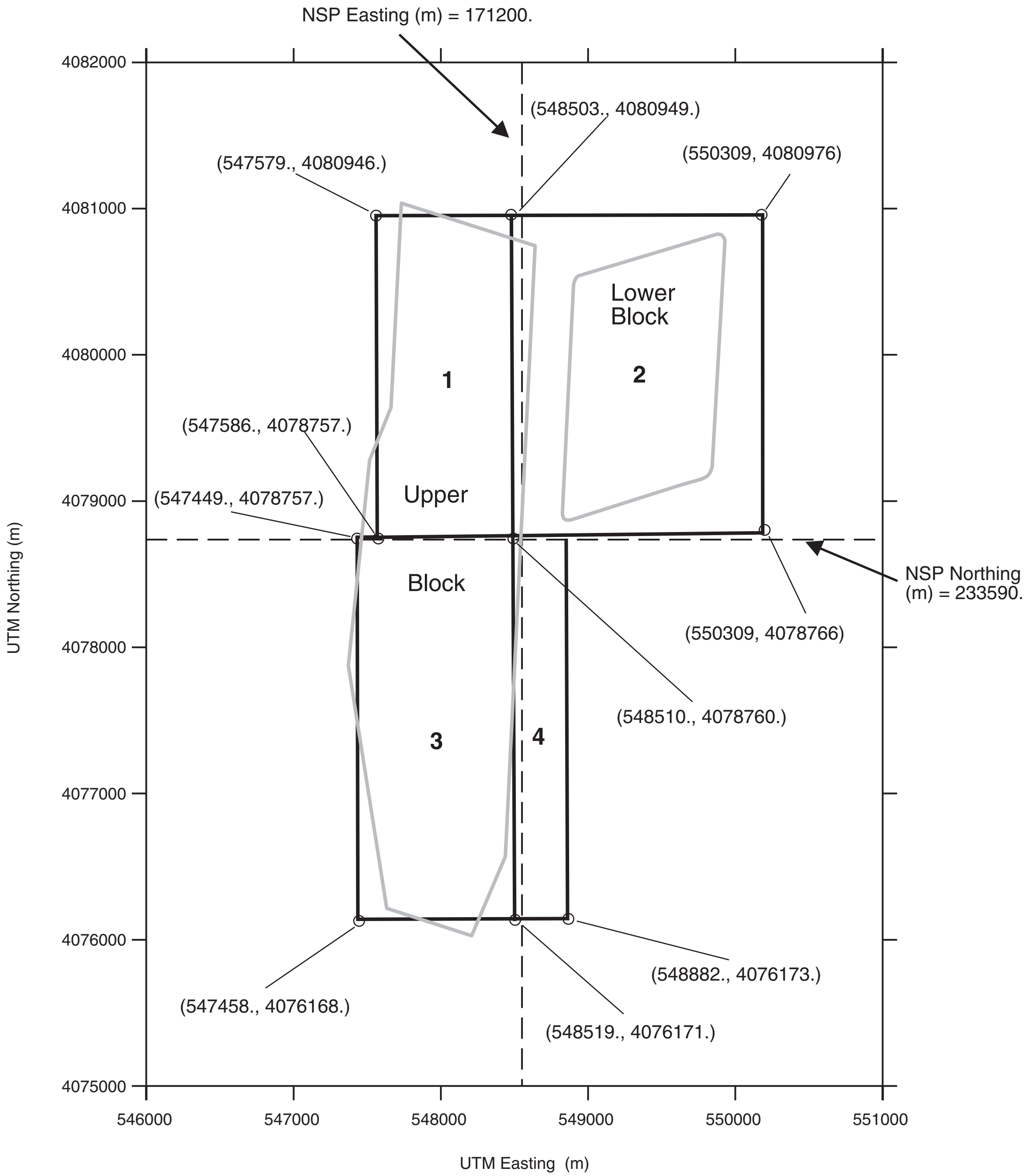
Figure I-3. Approximate configuration of the proposed Yucca Mountain Repository.



Note: Source regions are outlined with the solid black rectangles and numbered from 1 to 4. The coordinates of the vertices of the source region rectangles are given in Universal Transverse Mercator (UTM) coordinates (meters). The outline of the repository is shown by the gray lines. The dashed lines indicate the quadrants from which radionuclide arrivals from the unsaturated zone model are applied to the saturated zone source regions. Coordinates of the dashed lines are given in Nevada State Plan (NSP) coordinates (meters).

Source: Adapted from DIRS 157307-BSC 2001, Figure 5-3, p. 34.

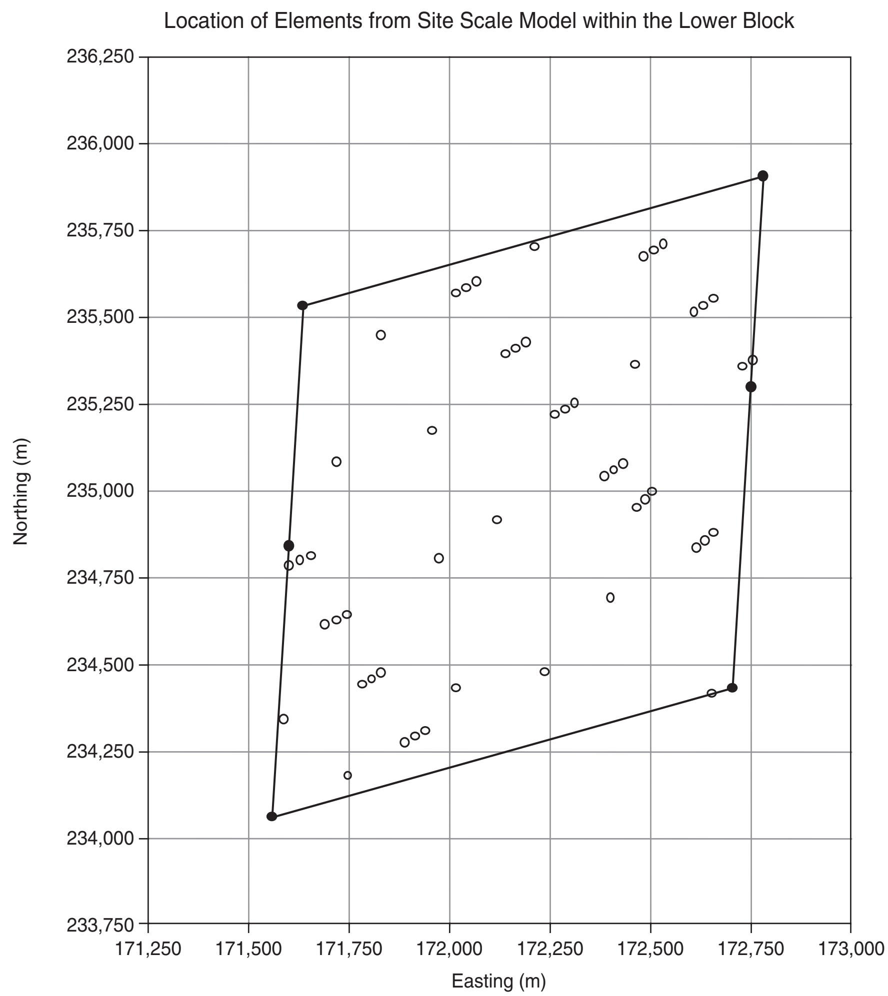
Figure I-4. The four saturated zone capture regions in relation to the primary repository block for the Proposed Action.



Note: Source regions are outlined with the solid black rectangles and numbered from 1 to 4. The coordinates of the vertices of the source region rectangles are given in Universal Transverse Mercator (UTM) coordinates (meters). The outline of the repository is shown by the gray lines. The dashed lines indicate the quadrants from which radionuclide arrivals from the unsaturated zone model are applied to the saturated zone source regions. Coordinates of the dashed lines are given in Nevada State Plan (NSP) coordinates (meters).

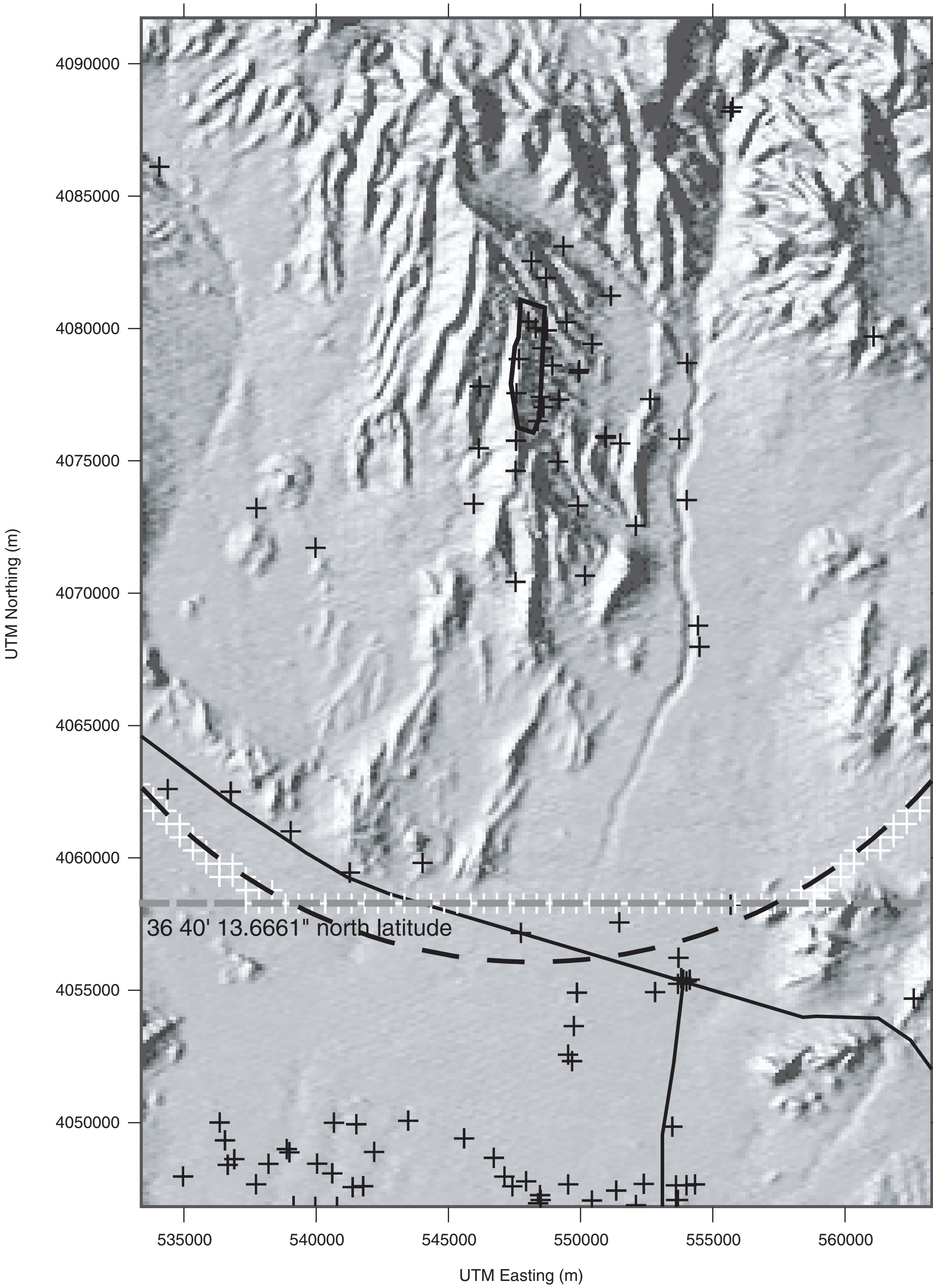
Source: Adapted from DIRS 157307-BSC 2001, Figure 5-4, p. I-35.

Figure I-5. The four saturated zone capture regions in relation to the primary and lower repository blocks for Inventory Modules 1 and 2.



Source: Adapted from DIRS 157307-BSC 2001, Figure 5-5, p. 36.

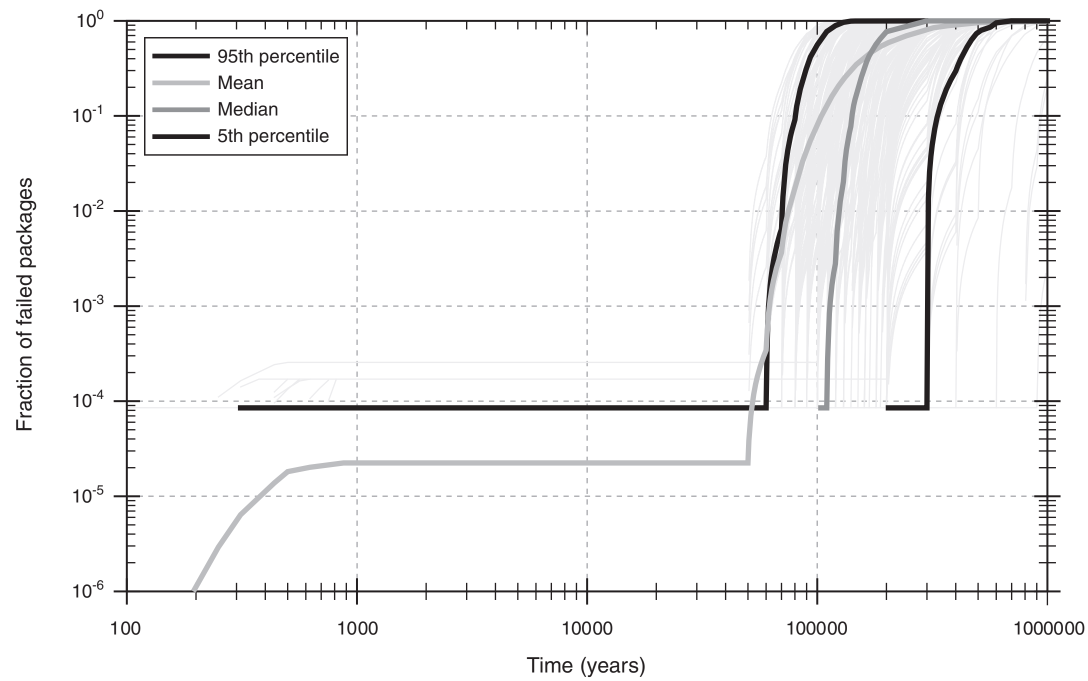
Figure I-6. Outline of the Lower Block showing the locations of the 51 particle-tracking nodes.



Note: The horizontal dashed line represents the southernmost boundary between the controlled area and accessible environment (36º 40' 13.6661" North latitude as per EPA 40 CFR Part 197). The semicircular dashed line represents the 20-kilometer (12-mile) fence used in the TSPA-SR model to evaluate compliance with the then-proposed Environmental Protection Agency and Nuclear Regulatory Commission regulations. The white crosses represent the saturated zone transport model grid cells at the intersection of the new southernmost boundary with the 20-kilometer fence used in TSPA-SR evaluations. The black crosses are well locations, and the solid black lines represent Highway 95 and Highway 373.

Source: Adapted from DIRS 157307-BSC 2001, Figure 5-6, p. 37.

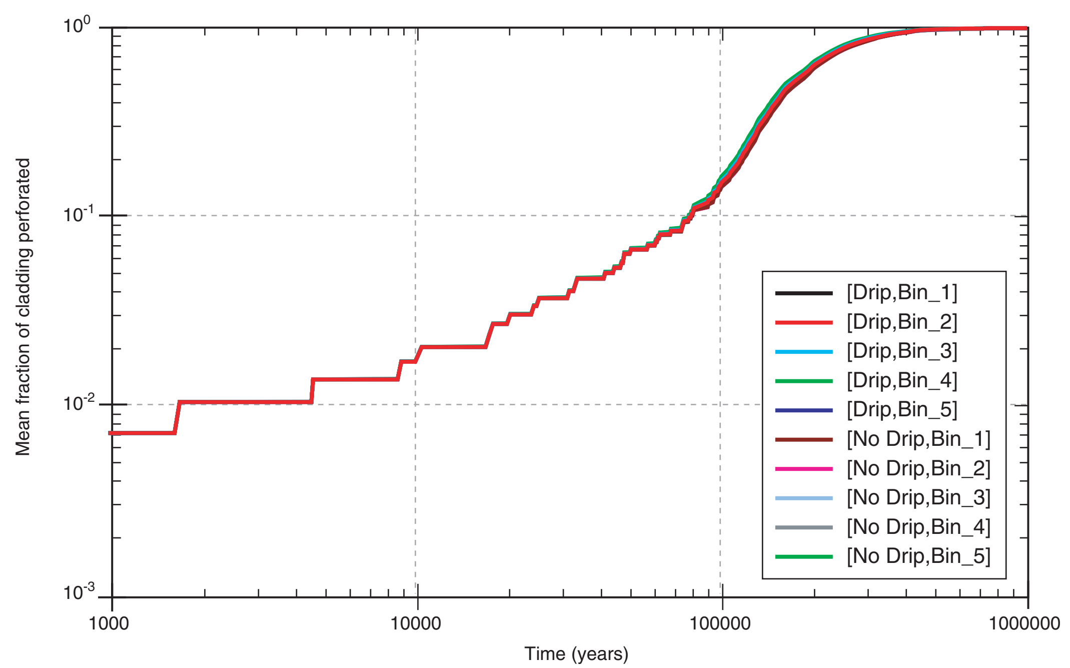
Figure I-7. Southernmost boundary of the controlled area and the accessible environment.



Note: Use of logarithmic scale on both axes.

Source: Adapted from DIRS 157307-BSC 2001, Figure 6-1, p. 38.

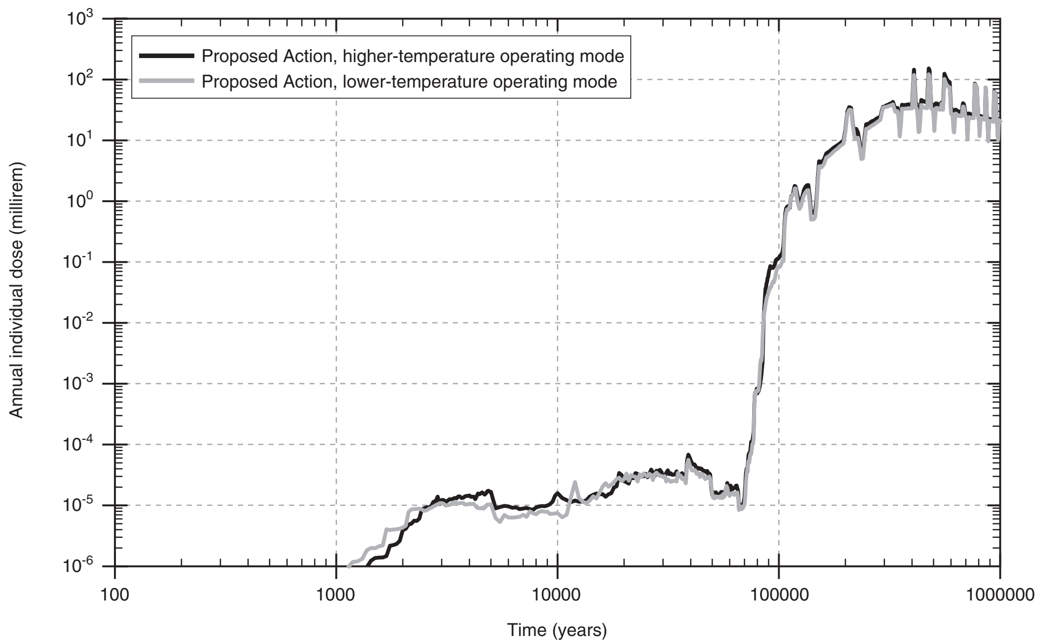
Figure I-8. Waste package failure curves for 300 probabilistic simulations for the Proposed Action inventory; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations.



Note: Use of logarithmic scale on both axes. All curves plot over one another when using a log scale for the y-axis.

Source: Adapted from DIRS 157307-BSC 2001, Figure 6-2, p. 39.

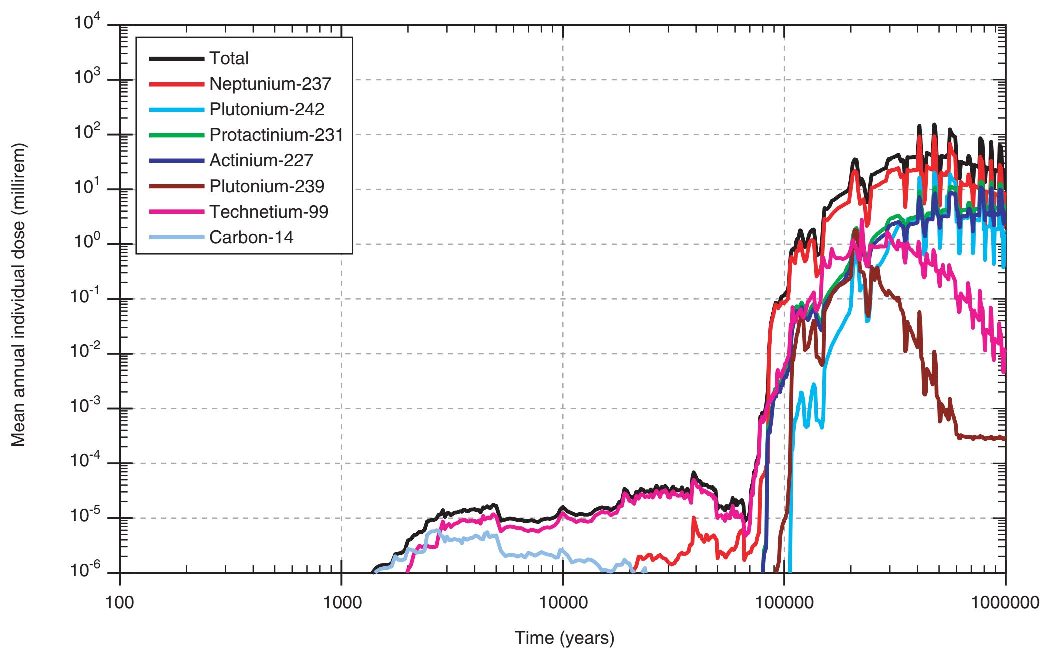
Figure I-9. Cladding failure profile for the Proposed Action inventory.



Note: Use of logarithmic scale on both axes.

Source: Adapted from DIRS 157307-BSC 2001, Figure 6-3, p. 40.

Figure I-10. Comparison plot of the total mean annual individual dose at the RMEI location under the higher-temperature and lower-temperature operating modes for the Proposed Action inventory, nominal scenario.



Note: Use of logarithmic scale on both axes.

Source: Adapted from DIRS 157307-BSC 2001, Figure 6-4, p. 41.

Figure I-11. Total and individual radionuclide mean annual dose to an individual at the RMEI location for the higher-temperature operating mode for the Proposed Action inventory, nominal scenario.

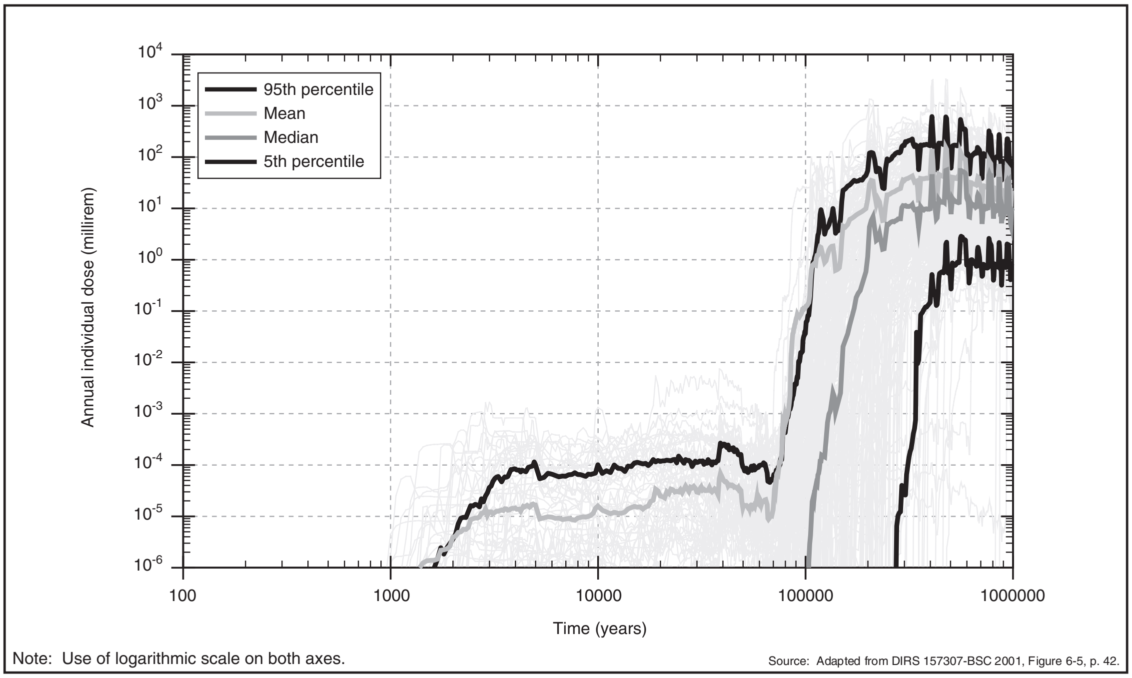


Figure I-12. Total annual individual dose at the RMEI location for 300 probabilistic simulations of the higher-temperature operating mode for the Proposed Action inventory, nominal scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations.

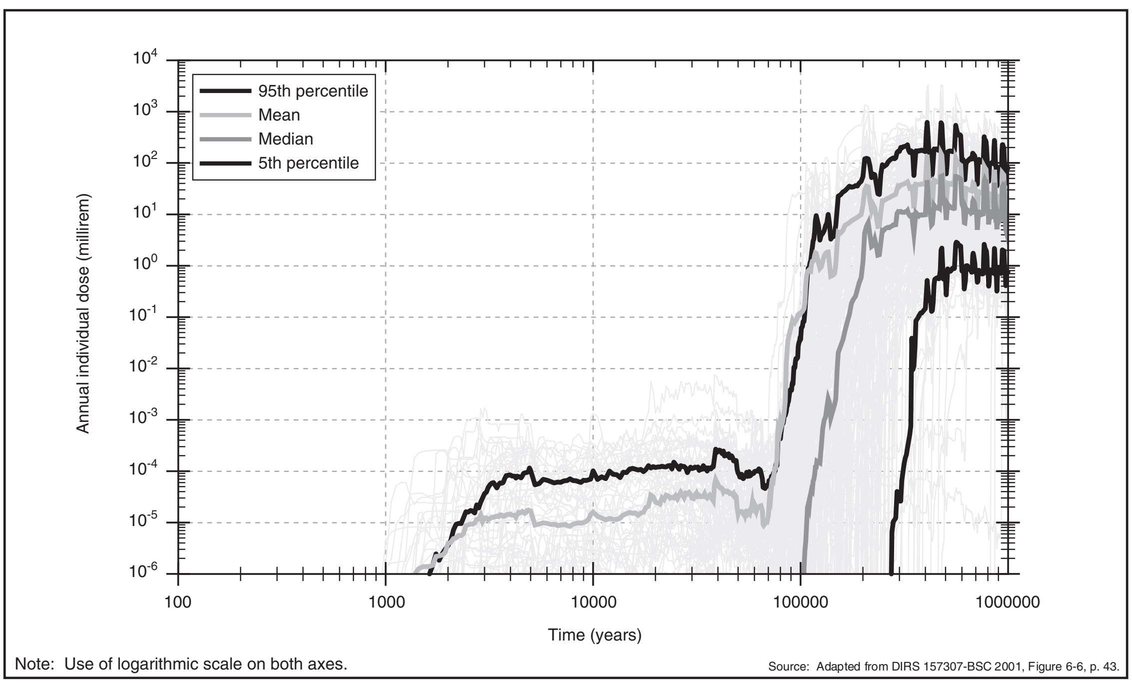


Figure I-13. Total annual individual dose at the RMEI location for 300 probabilistic simulations of the lower-temperature operating mode for the Proposed Action inventory, nominal scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations.

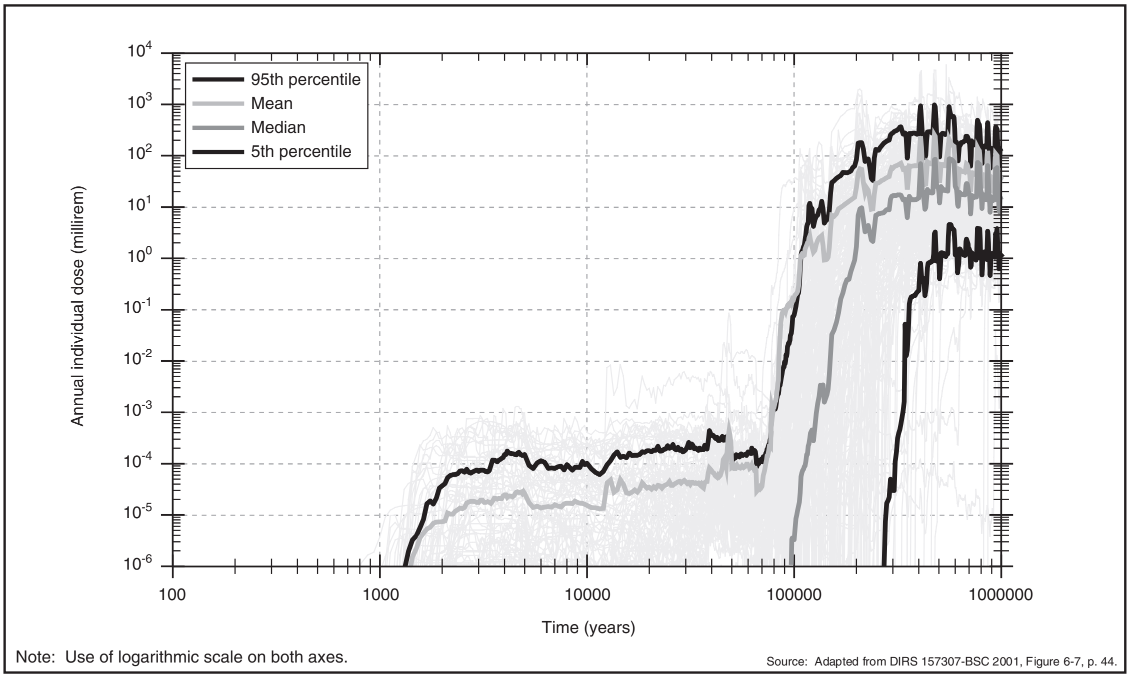


Figure I-14. Total annual individual dose at the RMEI location for 300 probabilistic simulations of the higher-temperature operating mode for the Module 1 inventory, nominal scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations.

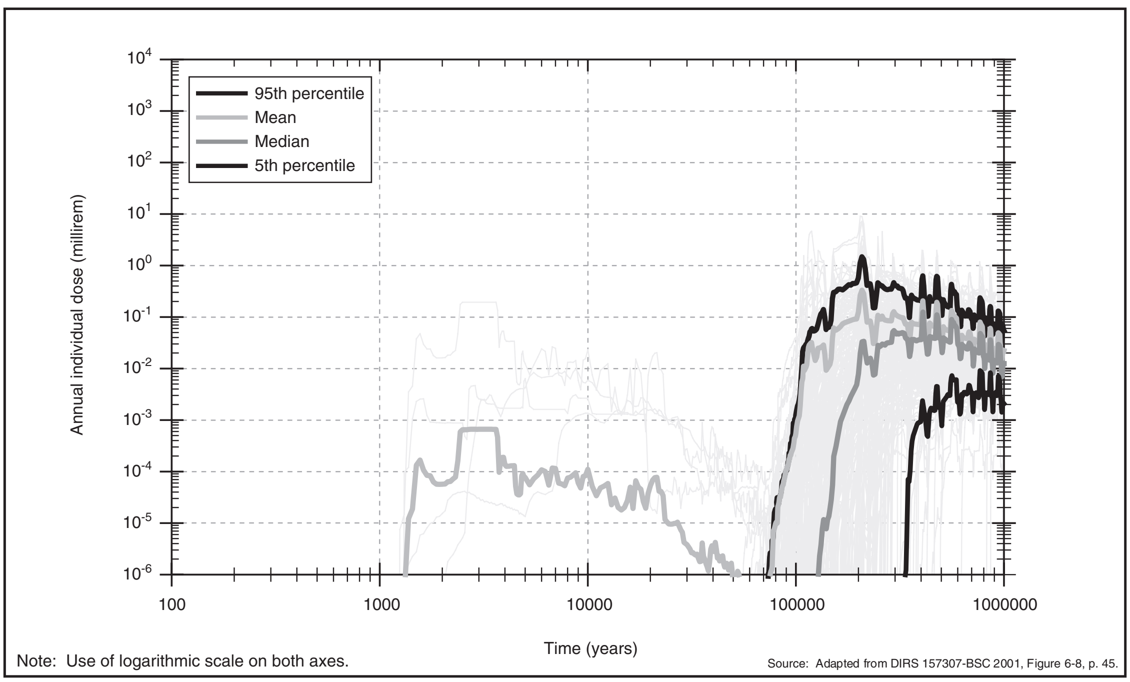


Figure I-15. Total annual individual dose at the RMEI location for 300 probabilistic simulations of the higher-temperature operating mode for the Module 2 incremental inventory, nominal scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations.

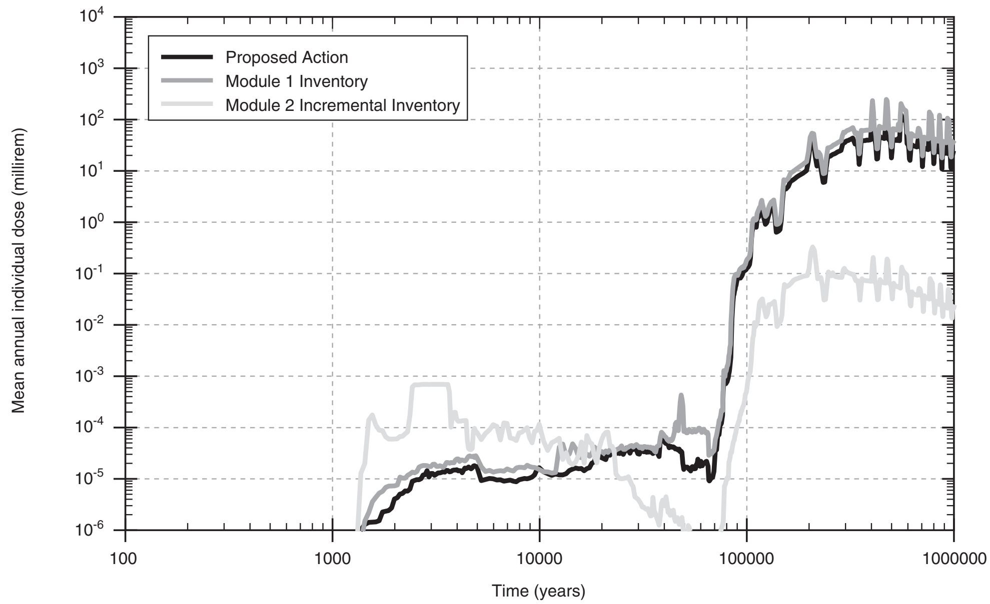


Figure I-16. Comparison plot of the mean total annual individual dose at the RMEI location for the higher-temperature operating mode for the Proposed Action, Module 1, and incremental Module 2 inventories, nominal scenario.

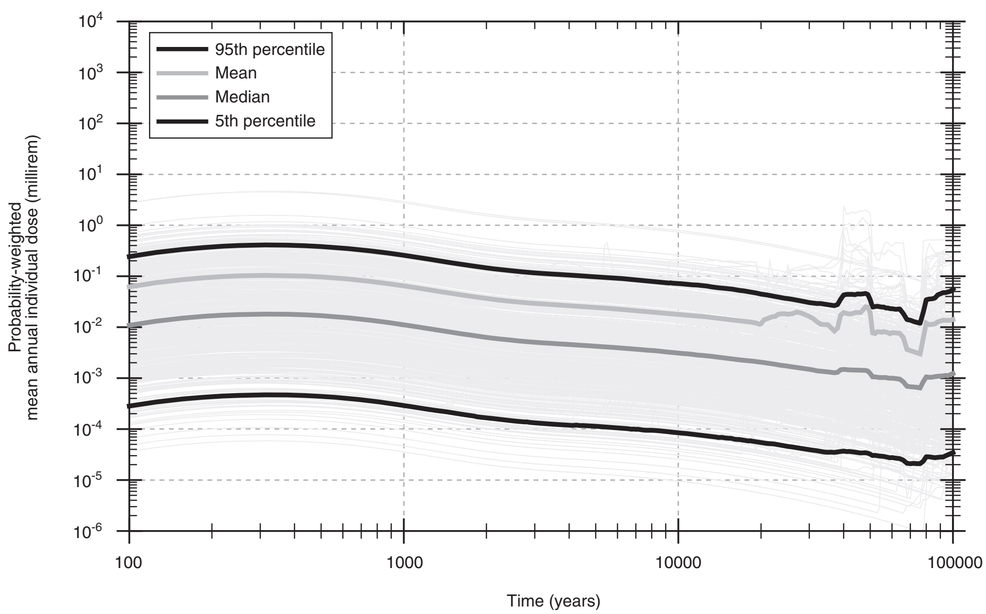


Figure I-17. Total annual individual dose at the RMEI location for 500 out of 5,000 probabilistic simulations of the higher-temperature operating mode for the Proposed Action inventory under the igneous activity scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of all 5,000 simulations.

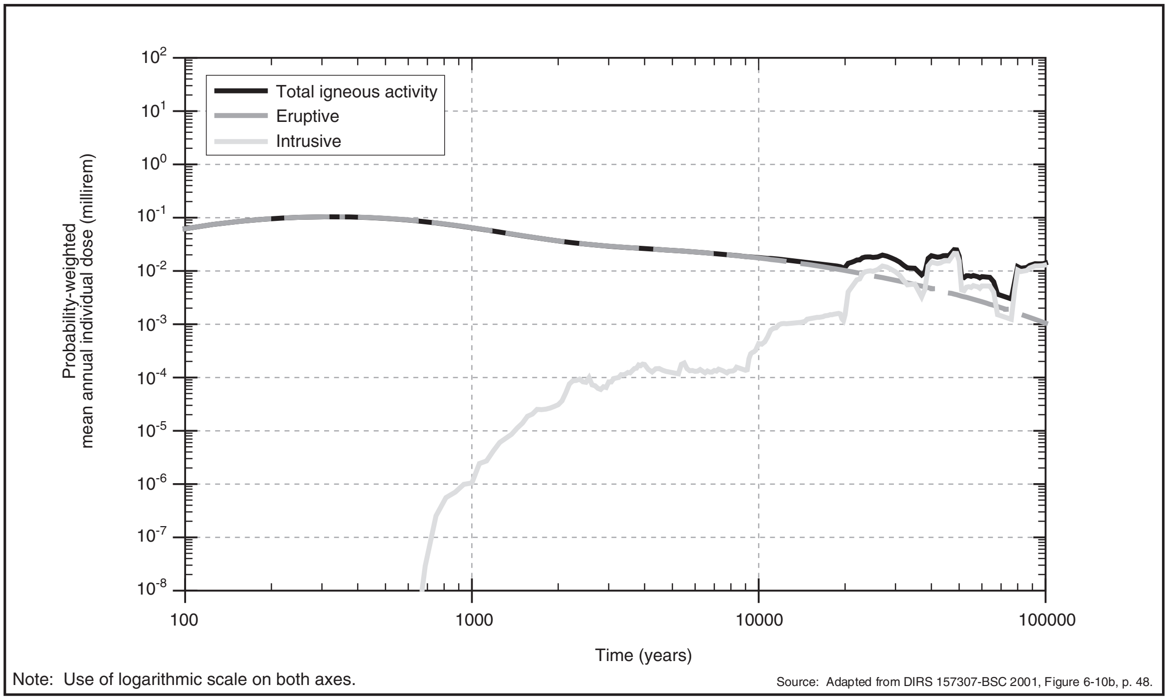


Figure I-18. Total mean individual receptor dose at the RMEI location for the higher-temperature operating mode for the Proposed Action inventory under the igneous activity scenario; the figure displays the mean results for both the eruptive and intrusive events and the sum of these events as "Total Igneous."

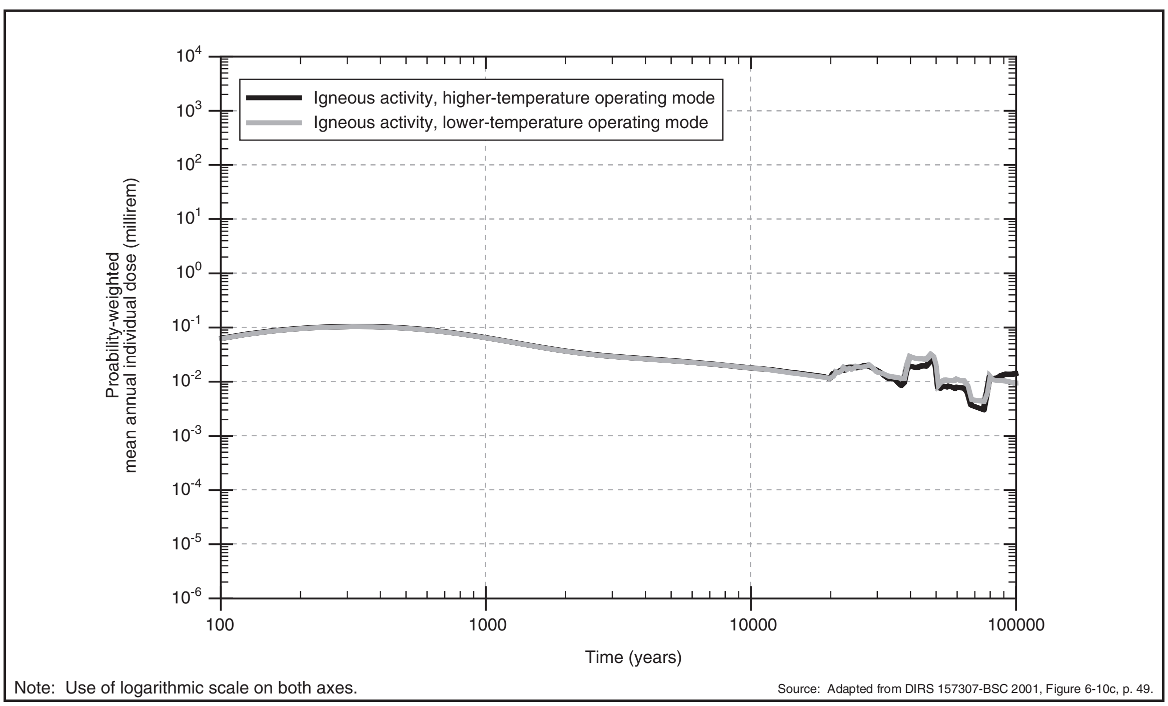


Figure I-19. Total mean annual individual dose at the RMEI location for the higher-temperature and lower-temperature operating modes for the Proposed Action inventory under the igneous activity scenario.

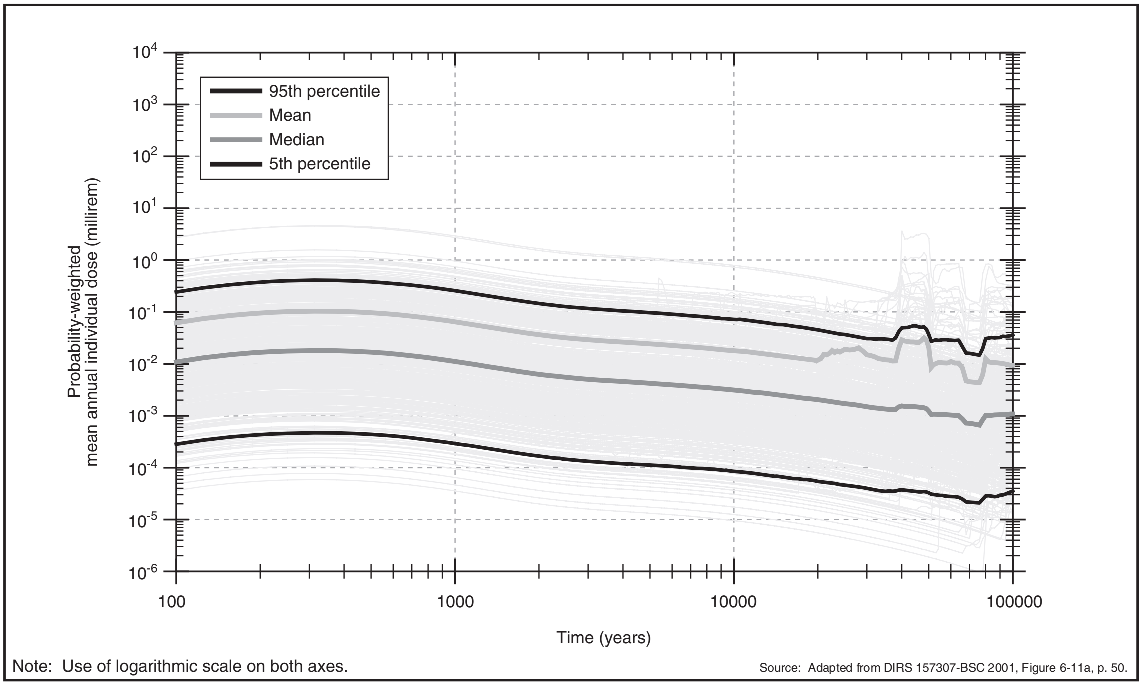


Figure I-20. Total annual individual dose at the RMEI location for 500 out of 5,000 probabilistic simulations of the lower-temperature operating mode for the Proposed Action inventory under the igneous activity scenario; the figure also displays the 5th-percentile, median, mean, and 95th-percentile values of these simulations.

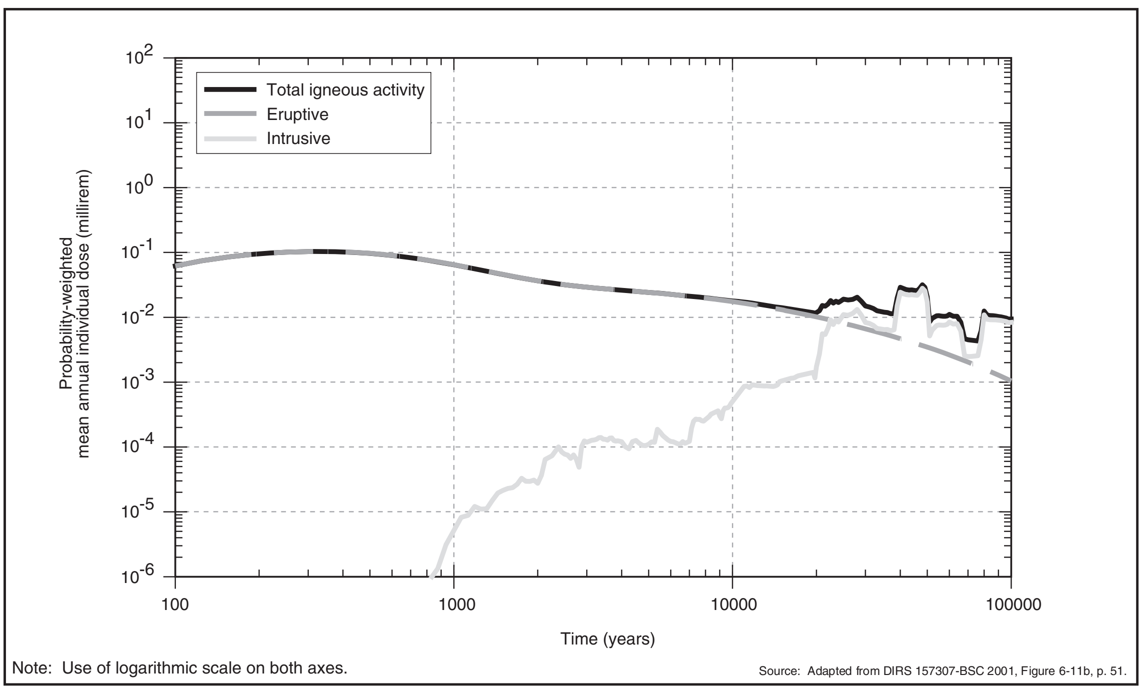
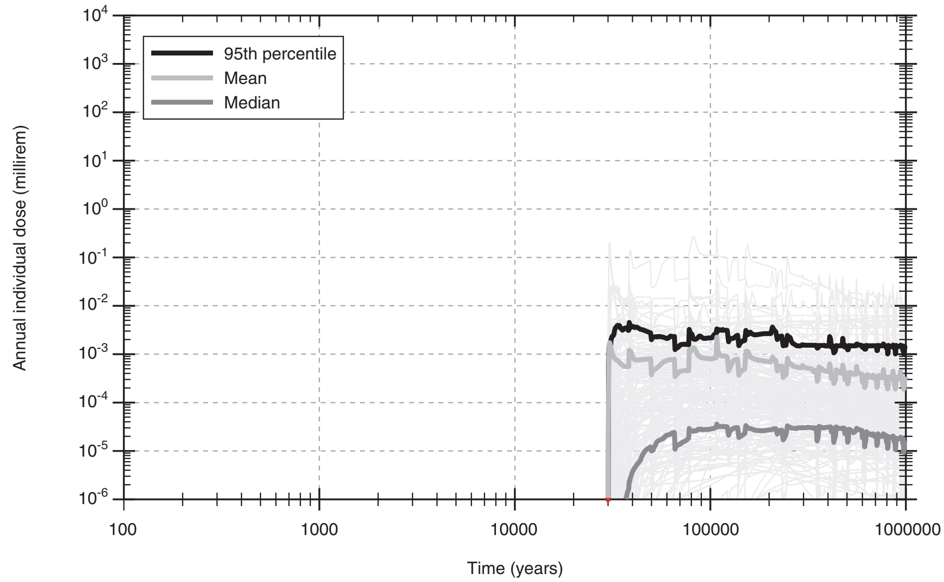


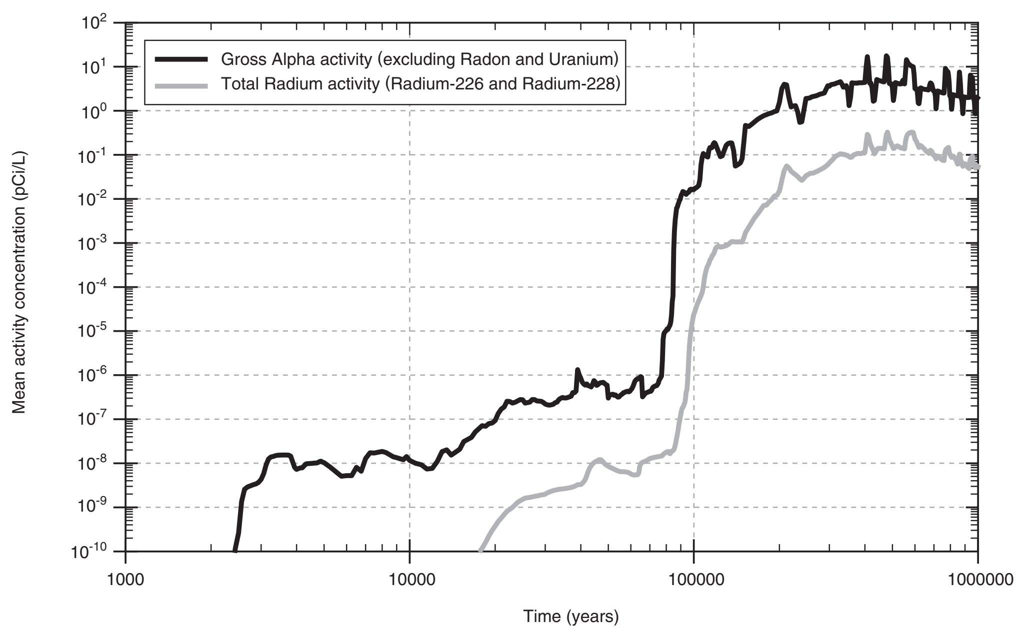
Figure I-21. Total mean annual individual dose at the RMEI location for the lower-temperature operating mode for the Proposed Action inventory under the igneous activity scenario; the figure displays the mean results for both the eruptive and intrusive events and the sum of these events as "Total Igneous."



Note: Use of logarithmic scale on both axes.

Source: Adapted from DIRS 157307-BSC 2001, Figure 6-12, p. 52.

Figure I-22. Total annual individual dose at the RMEI location for 300 probabilistic simulations of the higher-temperature operating mode for the Proposed Action inventory under the human intrusion-at-30,000-years scenario; the figure also displays the median, mean, and 95th-percentile values of these simulations.



Note: Use of logarithmic scale on both axes.

Source: Adapted from DIRS 157307-BSC 2001, Figure 6-17, p. 57.

Figure I-23. Mean activity concentrations of gross alpha activity and total radium (radium-226 plus radium-228) at the RMEI location of 300 probabilistic simulations of the higher-temperature operating mode for the Proposed Action inventory for the nominal scenario.

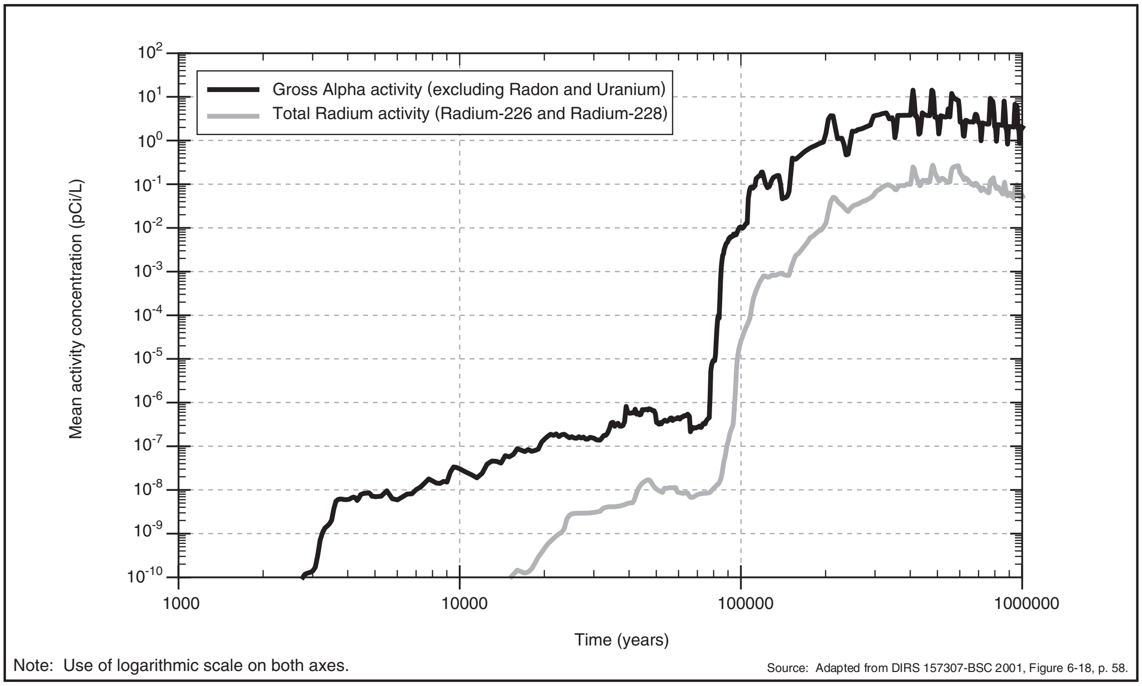


Figure I-24. Mean activity concentrations of gross alpha activity and total radium (radium-226 plus radium-228) at the RMEI location of 300 probabilistic simulations of the lower-temperature operating mode for the Proposed Action inventory for the nominal scenario.

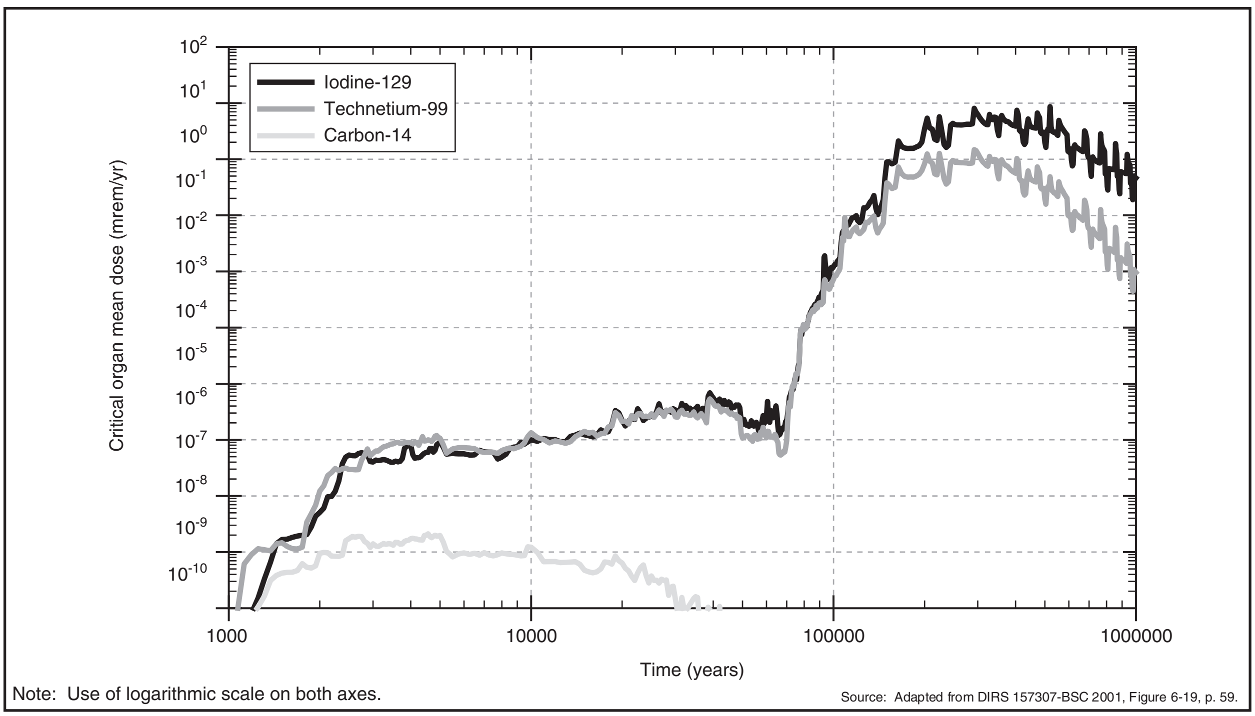


Figure I-25. Mean dose to critical organs for technetium-99, carbon-14, and iodine-129 at the RMEI location of 300 probabilistic simulations of the higher-temperature operating mode for the Proposed Action inventory for the nominal scenario.

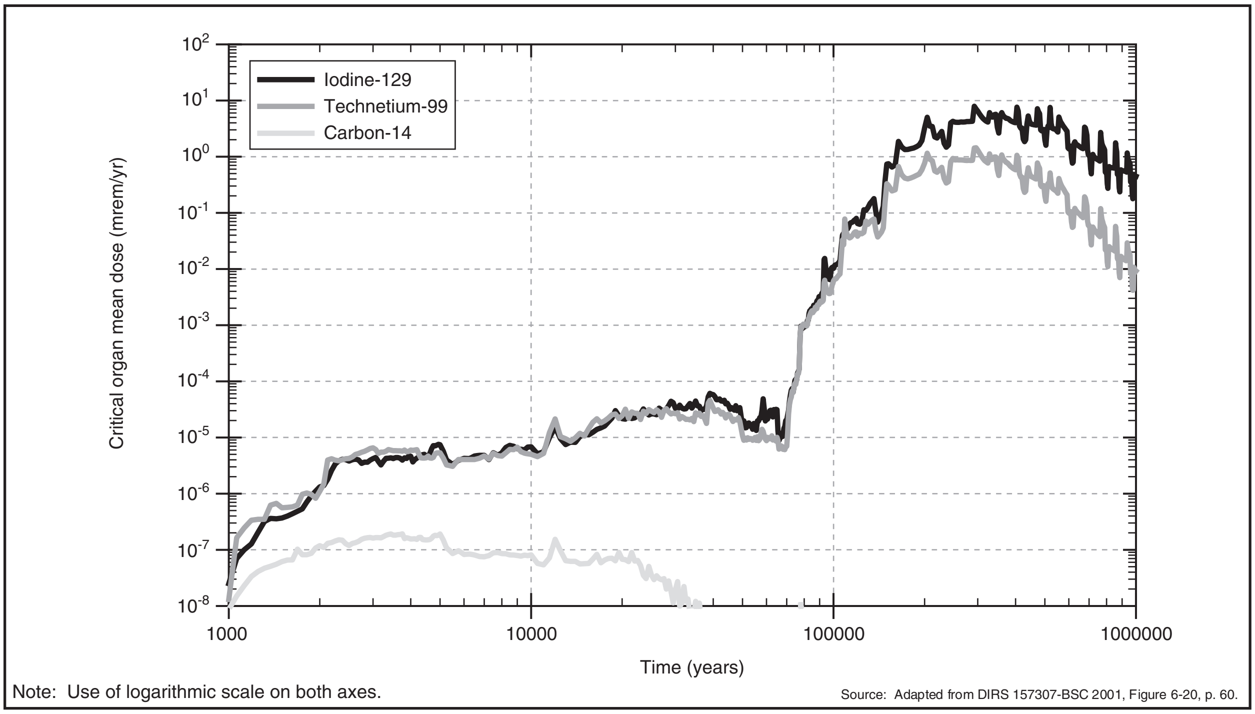


Figure I-26. Mean dose to critical organs for technetium-99, carbon-14, and iodine-129 at the RMEI location of 300 probabilistic simulations of the lower-temperature operating mode for the Proposed Action inventory for the nominal scenario.

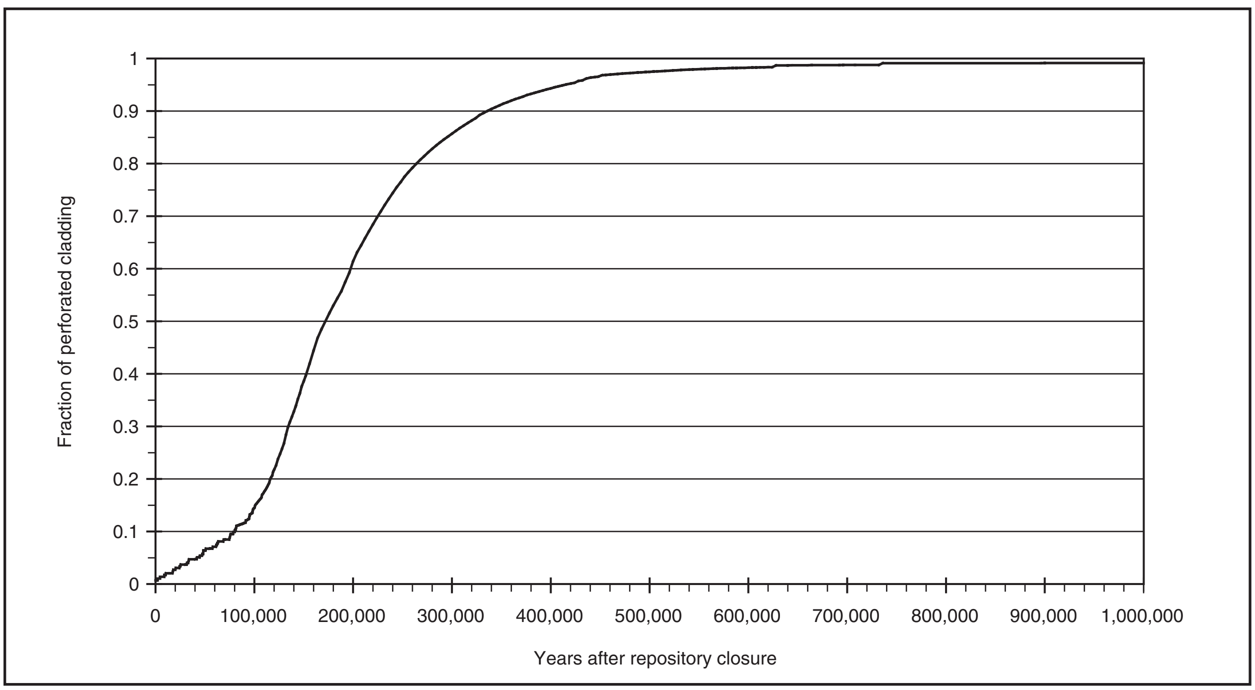


Figure I-27. Fraction of perforated cladding for commercial spent nuclear fuel as a function of time after repository closure.

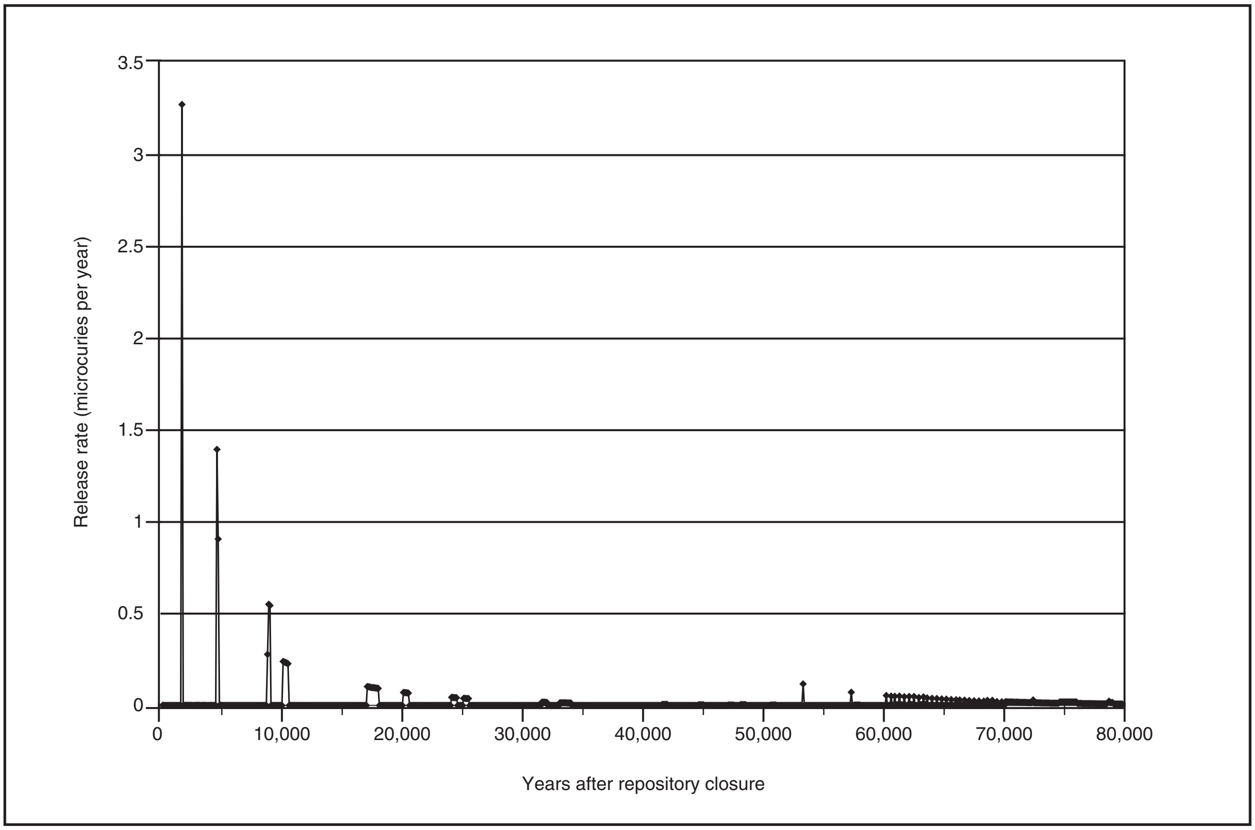


Figure I-28. Release rate of carbon-14 from the repository to the ground surface for 80,000 years following repository closure.

REFERENCES

Note: In an effort to ensure consistency among Yucca Mountain Project documents, DOE has altered the format of the references and some of the citations in the text in this Final EIS from those in the Draft EIS. The following list contains notes where applicable for references cited differently in the Draft EIS.

104328	ASTM 1998	ASTM (American Society for Testing and Materials) 1998. *Standard Specification for Low-Carbon Nickel-Molybdenum-Chromium, Low-Carbon Nickel-Chromium-Molybdenum, Low-Carbon Nickel-Chromium-Molybdenum-Copper and Low-Carbon Nickel-Chromium-Molybdenum-Tungsten Alloy Plate, Sheet, and Strip.* ASTM B 575-97. West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 241816.
100103	Bodvarsson, Bandurraga, and Wu 1997	Bodvarsson, G.S.; Bandurraga, T.M.; and Wu, Y.S., eds. 1997. *The Site-Scale Unsaturated Zone Model of Yucca Mountain, Nevada, for the Viability Assessment.* LBNL-40376. Berkeley, California: Lawrence Berkeley National Laboratory. ACC: MOL.19971014.0232.

152059	BSC 2001	BSC (Bechtel SAIC Company) 2001. *Performance Assessment of U.S. Department of Energy Spent Fuels in Support of Site Recommendation.* CAL-WIS-PA-000002 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010627.0026.
153909	BSC 2001	BSC (Bechtel SAIC Company) 2001. *Waste Package Radionuclide Inventory Approximations for TSPA-SR.* CAL-WIS-MD-000004 REV 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010227.0015.
154659	BSC 2001	BSC (Bechtel SAIC Company) 2001. *FY01 Supplemental Science and Performance Analyses, Volume 2: Performance Analyses.* TDR-MGR-PA-000001 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010724.0110.
154826	BSC 2001	BSC (Bechtel SAIC Company) 2001. *Features, Events, and Processes in UZ Flow and Transport.* ANL-NBS-MD-000001 REV 01. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010423.0321.
154841	BSC 2001	BSC (Bechtel SAIC Company) 2001. *Inventory Abstraction.* ANL-WIS-MD-000006 REV 00 ICN 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010416.0088.
155182	BSC 2001	BSC (Bechtel SAIC Company) 2001. *Software Code: GoldSim.* V7.17.200. 10344-7.17.200-00.
155950	BSC 2001	BSC (Bechtel SAIC Company) 2001. *FY01 Supplemental Science and Performance Analyses, Volume 1: Scientific Bases and Analyses.* TDR-MGR-MD-000007 REV 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010801.0404; MOL.20010712.0062; MOL.20010815.0001.
156016	BSC 2001	BSC (Bechtel SAIC Company) 2001. *Calculation: Consumption Rates of Locally Produced Food in Nye and Lincoln Counties.* CAL-MGR-EV-000002 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010628.0242.
157307	BSC 2001	BSC (Bechtel SAIC Company) 2001. *Total System Performance Assessment-Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain—Input to Final Environmental Impact Statement and Site Suitability Evaluation.* REV 00 ICN 02. Las Vegas, Nevada: Bechtel SAIC Co., LLC. ACC: MOL.20011213.0056.
102877	CRWMS M&O 1999	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 1999. *Environmental Baseline File for Meteorology and Air Quality.* B00000000-01717-5705-00126 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990302.0186. In the Draft EIS, this reference was cited as TRW 1999g in Chapter 12.

102933	CRWMS M&O 1999	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 1999. *Waste Package Materials Properties.* BBA000000-01717-0210-00017 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990407.0172.
119348	CRWMS M&O 1999	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor). *1999 Design Basis Waste Input Report for Commercial Spent Nuclear Fuel.* B00000000-01717-5700-00041 REV 00. Washington, D.C.: CRWMS M&O. ACC: MOV.19991006.0003.
136428	CRWMS M&O 1999	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 1999. *BWR Source Term Generation and Evaluation.* BBAC00000-01717-0210-00006 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000113.0334.
136429	CRWMS M&O 1999	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 1999. *PWR Source Term Generation and Evaluation.* BBAC00000-01717-0210-00010 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000113.0333.
147072	CRWMS M&O 1999	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 1999. *Source Terms for HLW Glass Canisters.* CAL-MGR-NU-000002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000124.0244.
135968	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Abstraction of Models for Stainless Steel Structural Material Degradation.* ANL-EBS-PA-000005 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000526.0337.
136281	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Evaluate Soil/Radionuclide Removal by Erosion and Leaching.* ANL-NBS-MD-000009 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000310.0057.
136383	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Inventory Abstraction.* ANL-WIS-MD-000006 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000414.0643.
138239	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Waste Packages and Source Terms for the Commercial 1999 Design Basis Waste Streams.* CAL-MGR-MD-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000214.0479.

139440	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Input and Results of the Base Case Saturated Zone Flow and Transport Model for TSPA.* ANL-NBS-HS-000030 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000526.0330.
139610	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Multiscale Thermohydrologic Model.* ANL-EBS-MD-000049 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000609.0267.
148384	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Total System Performance Assessment (TSPA) Model for Site Recommendation.* MDL-WIS-PA-000002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001226.0003.
149005	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Per Canister Inventories for DOE SNF for TSPA-SR.* CAL-WIS-MD-000006 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000510.0155.
149862	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Multiscale Thermohydrologic Model.* ANL-EBS-MD-000049 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001208.0062.
150558	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Update to the EIS Engineering File for the Waste Package in Support of the Final EIS.* TDR-EBS-MD-000010 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000317.0446.
150941	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *FEIS Update to Engineering File - Subsurface Repository.* TDR-EBS-MD-000007 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000612.0058.
151109	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Draft TSPA-SR Model Document.* Input Transmittal 00325.T. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000725.0018.
151551	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Characterize Framework for Igneous Activity at Yucca Mountain, Nevada.* ANL-MGR-GS-000001 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001221.0001.

151566	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *WAPDEG Analysis of Waste Package and Drip Shield Degradation.* ANL-EBS-PA-000001 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001208.0063.
151853	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Monitored Geologic Repository Project Description Document.* TDR-MGR-SE-000004 REV 02. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001031.0062.
151945	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Yucca Mountain Site Description.* TDR-CRW-GS-000001 REV 01 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001003.0111.
152542	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Calculation of General Corrosion Rate of Drip Shield and Waste Package Outer Barrier to Support WAPDEG Analysis.* CAL-EBS-PA-000002 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001024.0075.
153038	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Documentation of Million-Year TSPA.* Input Transmittal 00393.T. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001110.0057; MOL.20001120.0173.
153246	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Total System Performance Assessment for the Site Recommendation.* TDR-WIS-PA-000001 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001220.0045.
153596	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Radioactive Decay and In-Growth Modeling Approximations for TSPA-SR.* CAL-WIS-MD-000003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001226.0507.
153597	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *Relative Contribution of Individual Radionuclides to Inhalation and Ingestion Dose – One Million Years.* CAL-WIS-MD-000005 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20010103.0209.
155393	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2000. *EIS Performance-Assessment Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain.* CAL-CRW-PA-000002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000801.0002.

152539	CRWMS M&O 2001	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2001. *Nominal Performance Biosphere Dose Conversion Factor Analysis.* ANL-MGR-MD-000009 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20010123.0123.
153931	CRWMS M&O 2001	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2001. *Features, Events, and Processes in SZ Flow and Transport.* ANL-NBS-MD-000002 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20010214.0230.
154594	CRWMS M&O 2001	CRWMS M&O (Civilian Radioactive Waste Management System Management & Operating Contractor) 2001. *Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux.* ANL-EBS-HS-000003 REV 00 ICN 02. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20010221.0160.
102588	DOE 1992	DOE (U.S. Department of Energy) 1992. *Characteristics of Potential Repository Wastes.* DOE/RW-0184-R1. Four volumes. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: HQO.19920827.0001; HQO.19920827.0002; HQO.19920827.0003; HQO.19920827.0004.
101798	DOE 1994	DOE (U.S. Department of Energy) 1994. *Greater-Than-Class C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities, and Other Characteristics.* DOE/LLW-114, Rev. 1. Idaho Falls, Idaho: U.S. Department of Energy. TIC: 231330.
100332	DOE 1997	DOE (U.S. Department of Energy) 1997. *The 1997 "Biosphere" Food Consumption Survey Summary Findings and Technical Documentation.* Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19981021.0301.
101779	DOE 1998	DOE (U.S. Department of Energy) 1998. *Viability Assessment of a Repository at Yucca Mountain. DOE/RW-0508. Overview and five volumes.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19981007.0027; MOL.19981007.0028; MOL.19981007.0029; MOL.19981007.0030; MOL.19981007.0031; MOL.19981007.0032.
102205	DOE 1998	DOE (U.S. Department of Energy) 1998. *Guide of Good Practices for Occupational Radiological Protection in Plutonium Facilities.* Washington, D.C.: U.S. Department of Energy. ACC: MOL.20010721.0029.
153849	DOE 2001	DOE (U.S. Department of Energy) 2001. *Yucca Mountain Science and Engineering Report.* DOE/RW-0539. [Washington, D.C.]: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20010524.0272.

103705	EPA 1997	EPA (U.S. Environmental Protection Agency) 1997. *Health Effects Assessment, Summary Tables, FY-1997 Update.* EPA 540/R-97-036. Washington, D.C.: U.S. Environmental Protection Agency. TIC: 243784.
101825	EPA 1998	EPA (U.S. Environmental Protection Agency) 1998. *Toxicological Review of Hexavalent Chromium [Chromium (VI)].* CAS No. 18540-29-9. Washington, D.C.: U.S. Environmental Protection Agency. TIC: 243147.
148219	EPA 1999	EPA (U.S. Environmental Protection Agency) 1999. "Boron (Boron and Borates only); CARSN 7440-42-8." IRIS (Integrated Risk Information System). Washington, D.C.: U.S. Environmental Protection Agency. Accessed June 10, 1999. TIC: 244100. http://www.epa.gov/iris/subst/0410.htm
148221	EPA 1999	EPA (U.S. Environmental Protection Agency) 1999. "Cadmium; CASRN 7440-43-9." IRIS (Integrated Risk Information System). Washington, D.C.: U.S. Environmental Protection Agency. Accessed June 10, 1999. TIC: 244101. http://www.epa.gov/iris/subst/0141.htm
148224	EPA 1999	EPA (U.S. Environmental Protection Agency) 1999. "Chromium (VI); CASRN 18540-29-9." IRIS (Integrated Risk Information System). Washington, D.C.: U.S. Environmental Protection Agency. Accessed June 10, 1999. TIC: 244103. http://www.epa.gov/iris/subst/0144.htm
148227	EPA 1999	EPA (U.S. Environmental Protection Agency) 1999. "Manganese; CASRN 7439-96-5." IRIS (Integrated Risk Information System). Washington, D.C.: U.S. Environmental Protection Agency. Accessed June 10, 1999. TIC: 244104. http://www.epa.gov/iris/subst/0373.htm
148228	EPA 1999	EPA (U.S. Environmental Protection Agency) 1999. "Molybdenum; CASRN 7439-98-7." IRIS (Integrated Risk Information System). Washington, D.C.: U.S. Environmental Protection Agency. Accessed June 10, 1999. TIC: 244105. http://www.epa.gov/iris/subst/0425.htm
148229	EPA 1999	EPA (U.S. Environmental Protection Agency) 1999. "Nickel, Soluble Salts; CASRN Various." IRIS (Integrated Risk Information System). Washington, D.C.: U.S. Environmental Protection Agency. Accessed June 10, 1999. TIC: 244108. http://www.epa.gov/iris/subst/0421.htm
148233	EPA 1999	EPA (U.S. Environmental Protection Agency) 1999. "Zinc and Compounds; CASRN 7440-66-6." IRIS (Integrated Risk Information System). Washington, D.C.: U.S. Environmental Protection Agency. Accessed June 10, 1999. TIC: 224107. http://www.epa.gov/iris/subst/0426.htm
100033	Flint 1998	Flint, L.E. 1998. *Characterization of Hydrogeologic Units Using Matrix Properties, Yucca Mountain, Nevada.* Water-Resources Investigations Report 97-4243. Denver, Colorado: U.S. Geological Survey. ACC: MOL.19980429.0512.

155937	Freeze 2001	Freeze, G. 2001. "EIS Post-10,000 Year FEP's." E-mail from G. Freeze (Duke) to D. Lester (Jason Technologies), September 10, 2001. ACC: MOL.20011009.0061.
154365	Freeze, Brodsky, and Swift 2001	Freeze, G.A.; Brodsky, N.S.; and Swift, P.N. 2001. *The Development of Information Catalogued in REV00 of the YMP FEP Database.* TDR-WIS-MD-000003 REV 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010301.0237.
118942	Fridrich 1999	Fridrich, C.J. 1999. "Tectonic Evolution of the Crater Flat Basin, Yucca Mountain Region, Nevada." Chapter 7 of *Cenozoic Basins of the Death Valley Region.* Wright, L.A. and Troxel, B.W., eds. Special Paper 333. Boulder, Colorado: Geological Society of America. TIC: 248054.
151395	Golder Associates 1999	Golder Associates 1999. *Software Code: RIP.* 5.19.01. 30055 V5.19.01.
100814	Harrar et al. 1990	Harrar, J.E.; Carley, J.F.; Isherwood, W.F.; and Raber, E. 1990. *Report of the Committee to Review the Use of J-13 Well Water in Nevada Nuclear Waste Storage Investigations.* UCID-21867. Livermore, California: Lawrence Livermore National Laboratory. ACC: NNA.19910131.0274.
101074	ICRP 1975	ICRP (International Commission on Radiological Protection) 1975. *Report of the Task Group on Reference Man: A Report Prepared by a Task Group of Committee 2 of the International Commission on Radiological Protection.* ICRP Number 23. Tarrytown, New York: Elsevier Science. TIC: 237218.
110431	INEEL 1999	INEEL (Idaho National Engineering and Environmental Laboratory) 1999. *CD-Design Basis Event Data (DBE), Revision 1, National Spent Nuclear Fuel Program.* Idaho Falls, Idaho: Idaho National Engineering and Environmental Laboratory. ACC: MOL.19990225.0479.
148066	Killough and Rohwer 1978	Killough, G.G. and Rohwer, R.S. 1978. "A New Look at the Dosimetry of 14C Released to the Atmosphere as Carbon Dioxide." *Health Physics, 34,* 141-159. London, Great Britain: Pergamon Press. TIC: 243510.
103178	Lide and Frederikse 1997	Lide, D.R. and Frederikse, H.P.R., eds. 1997. *CRC Handbook of Chemistry and Physics.* 78th Edition. Boca Raton, Florida: CRC Press. TIC: 243741. In the Draft EIS this reference was cited as CRC 1997 in Appendix I.
100953	Napier et al. 1988	Napier, B.A.; Peloquin, R.A.; Strenge, D.L.; and Ramsdell, J.V. 1988. *Conceptual Representation.* Volume 1 of *GENII - The Hanford Environmental Radiation Dosimetry Software System.* PNL-6584. Richland, Washington: Pacific Northwest Laboratory. TIC: 206898.

103446	Oversby 1987	Oversby, V.M. 1987. "Spent Fuel as a Waste Form – Data Needs to Allow Long Term Performance Assessment under Repository Disposal Conditions." *Scientific Basis for Nuclear Waste Management X, Symposium held December 1-4, 1986, Boston, Massachusetts.* Bates, J.K. and Seefeldt, W.B., eds. 84, 87-101. Pittsburgh, Pennsylvania: Materials Research Society. TIC: 203663.
104418	Rowland 1997	Rowland, T.J. 1997. "Repository Environmental Impact Statement Data Call for High-Level Waste." Memorandum from T.J. Rowland (DOE) to K.G. Picha (DOE), November 26, 1997, JLD:04-53159-435.5, with attachment. ACC: MOL.19990608.0048.
118952	Savage, Svarc, and Prescott 1999	Savage, J.C.; Svarc, J.L.; and Prescott, W.H. 1999. "Strain Accumulation at Yucca Mountain, Nevada, 1983-1998." *Journal of Geophysical Research, 104,* (B8), 17627-17631. Washington, D.C.: American Geophysical Union. TIC: 245645.
100354	USGS 1998	USGS (U.S. Geological Survey) 1998. *Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada.* Milestone SP32IM3, June 15, 1998. Three volumes. Oakland, California: U.S. Geological Survey. ACC: MOL.19980619.0640.
136368	USGS 2000	USGS (U.S. Geological Survey) 2000. *Future Climate Analysis.* ANL-NBS-GS-000008 REV 00. Denver, Colorado: U.S. Geological Survey. ACC: MOL.20000629.0907.
127163	Weast 1972	Weast, R.C., ed. 1972. *CRC Handbook of Chemistry and Physics.* 53rd Edition. Cleveland, Ohio: Chemical Rubber Company. TIC: 219220.
157520	Williams 2001	Williams, N.H. 2001. "Contract No. DE-AC08-01RW12101-Corrections to Final Letter Report Describing the Performance of Proposed Yucca Mountain Repository for the Final Environmental Impact Statement and Site Suitability Evaluation." Letter from N.H. Williams (BSC) to J.R. Summerson (DOE/YMSCO), September 28, 2001, RWA:cs, with enclosure. ACC: MOL.200111114.0256.
100836	Wolery 1992	Wolery, T.J. 1992. *EQ3NR, A Computer Program for Geochemical Aqueous Speciation-Solubility Calculations: Theoretical Manual, User's Guide, and Related Documentation (Version 7.0).* UCRL-MA-110662 PT III. Livermore, California: Lawrence Livermore National Laboratory. TIC: 205154.
103690	Wu, Chen, and Bodvarsson 1995	Wu, Y.S.; Chen, G.; and Bodvarsson, G. 1995. *Preliminary Analysis of Effects of Thermal Loading on Gas and Heat Flow Within the Framework of the LBNL/USGS Site-Scale Model.* LBL- 37729. Berkeley, California: Lawrence Berkeley National Laboratory. TIC: 222270.

100520 YMP 1993 YMP (Yucca Mountain Site Characterization Project) 1993. *Evaluation of the Potentially Adverse Condition "Evidence of Extreme Erosion During the Quaternary Period" at Yucca Mountain, Nevada.* Topical Report YMP/92-41-TPR. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: NNA.19930316.0208.