TRW Environmental Safety Systems Inc.

Repository Safety Strategy:

Plan To Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations

TDR-WIS-RL-000001 REV 04 ICN 01

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Civilian Radioactive Waste Management System

Management & Operating Contractor

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Civilian Radioactive Waste Management System
Management & Operating Contractor

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November 2000

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11/21/00
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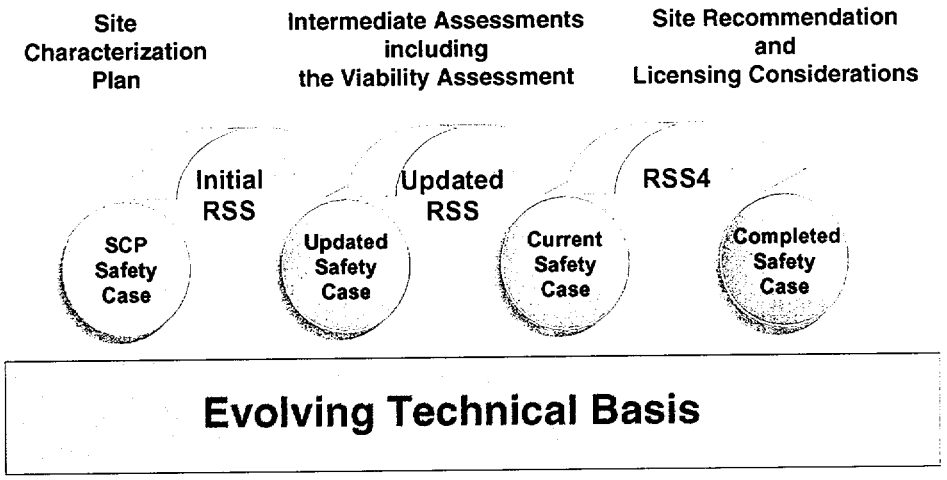
PREFACE

A key issue for the site recommendation and license application considerations for a high-level waste repository at Yucca Mountain is the ability of that site to protect the public from any undue radiological risk before and after permanent closure. The U.S. Department of Energy is therefore preparing a case for preclosure and postclosure safety of a potential repository system at Yucca Mountain. This document, Revision 4 of the Repository Safety Strategy (RSS), reports the current status of this safety case and the plans to complete it.

The sequence of decisions affecting site recommendation and licensing of a Yucca Mountain repository has been and will be made in light of increasing site information and design development. Accordingly, the safety case supporting those decisions evolves; the RSS therefore includes plans for that evolution. The general picture of the evolution of the safety case and the RSS is shown in the following figure.



Evolution of the Safety Case and the Repository Safety Strategy

This figure indicates the RSS at various stages. The initial version of the RSS was prepared as part of DOE's site characterization plan (SCP) for the Yucca Mountain site. A safety case, including an initial performance assessment, was prepared based upon preliminary site and design information. That safety case provided the basis for the plans for site investigations to address remaining issues. As information has been acquired and repository design developed, the safety case has been updated. Accordingly, the repository safety strategy has been revised.

This document presents Revision 4 of the RSS (RSS4). This revision presents the current status of the safety case and the plans for additional work to complete the case to support the site recommendation and licensing considerations. The current safety case is based

upon the information developed up to this time in the evolving process. Although system design is not yet complete and site investigations to address particular objectives identified in previous versions of the strategy are not yet finished, the safety case appears to be converging. A few issues remain for the site recommendation and licensing considerations and this revision provides the plans to address them.

As in previous versions, Revision 4 of the RSS focuses on radiological safety as defined in the regulatory performance objectives. These performance objectives have also evolved as the Nuclear Regulatory Commission has developed its regulations for a repository system. An important part of this regulatory evolution has been the adoption of a risk-informed, performance based approach to compliance with the performance objectives. This revision of the RSS fully subscribes to this approach.

The regulatory performance objectives address both preclosure and postclosure safety. Previous versions of the RSS addressed only postclosure safety. However, Revision 4 presents both a preclosure safety strategy and a postclosure safety strategy. These strategies are documented in Volumes I and II of the RSS, respectively. Because the preclosure and postclosure performance objectives are independent of one another, each volume stands alone. Each volume provides its own table of contents and glossary and is preceded by its own executive summary. However, the strategies reflect a common risk-informed, performance-based conceptual framework and are, in this sense, fully integrated.

CHANGE HISTORY

REV	ICN	Document No.	Date	Description Of Change
00	00	None	July 1996	Initial update of top-level strategy of site characterization plan to reflect results of site characterization.
01	00	YMP/96-01 REV 01	January 1998	Update strategy to reflect additional site characterization information and improved performance assessment models.
02	00	YMP/96-01 REV 02	December 1998	Update strategy to reflect latest performance assessment results and address design options and alternatives.
03	00	TDR-WIS-RL-000001 REV 03	January 2000	Update strategy to reflect latest performance assessment results and addressed enhanced design.
04	00	TDR-WIS-RL-000001 REV 04	October 2000	Update strategy to reflect latest performance assessment results and address design for site recommendation consideration report. Incorporate preclosure safety strategy.
04	01	TDR-WIS-RL-000001 REV 04 ICN 01	November 2000	Interim change to (1) enhance integration of Volumes I and II of strategy documentation; (2) improve consistency with Site Recommendation Consideration Report; and (3) improve conformance with referenced reports and regulations. Changes indicated by change bars in right margin.

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QA: QA

Civilian Radioactive Waste Management System
Management & Operating Contractor

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Volume I: Preclosure Safety Strategy

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Volume I: Preclosure Safety Strategy

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EXECUTIVE SUMMARY
PRECLOSURE SAFETY STRATEGY

A key issue for the site recommendation and licensing considerations for a high-level waste repository at Yucca Mountain is the ability of that site to protect the public from any undue radiological risk before and after permanent closure. The U.S. Department of Energy is therefore preparing a case regarding preclosure and postclosure safety of a potential repository system at Yucca Mountain. This document reports the current status of this safety case and the plans to complete it.

This document presents Revision 4 of the Repository Safety Strategy and reflects project developments and information learned since previous revisions. This volume, Volume I, presents the preclosure safety strategy. Volume II presents the postclosure safety strategy.

The following are the elements of the repository preclosure safety case. These elements combine to provide confidence in repository preclosure safety:

- Integrated safety analysis (ISA)
- Margin and defense-in-depth
- Consequence analysis of beyond-design-basis events
- Commercial Nuclear Industry precedent and experience
- License specifications and surveillances.

The central preclosure safety strategy is to design, construct, and operate structures, systems, and components (SSCs) that are important to safety to perform intended safety functions during and following credible internal and external events and natural phenomena in a manner that the design basis event dose limits are not exceeded. This strategy is based on the repository preclosure operations functions: (1) receipt of waste; (2) transfer of waste into the disposal container; (3) sealing the disposal container; (4) transfer of the waste package to the emplacement drift; (5) emplacement; and (6) monitoring until repository closure. The safety strategy for each of these basic functions will be to design and operate SSCs important to safety so as to prevent adverse events and conditions if practical while at the same time applying mitigation measures if required to ensure compliance with the dose limits. The strategy for the preclosure operational period is to maximize the use of technology and concepts proven for the safe handling of radioactive wastes over many years in the commercial nuclear industry and to take advantage of commercial power industry licensing precedents. The types of internal events considered include fire, explosions, implosions, chemical contamination, flooding, releases of radioactive material, exposure to high radiation fields, and potential for nuclear criticality. External events involve natural phenomena, such as earthquakes and volcanism, and external man-made hazards, such as those posed by aircraft and nearby military/industrial facilities. The safety strategy for the first five of the repository preclosure operations functions and the list of safety features for each are discussed in this document. A detailed safety assessment for the monitoring period has yet to be performed; however, it is anticipated that the relevant DBEs, safety strategy, safety features, and calculated offsite exposures will be bounded by those determined from evaluation of the first five functions.

Proposed 10 CFR Part 63 requires that compliance with the Nuclear Regulatory Commission's repository preclosure performance objectives for the Yucca Mountain repository be demonstrated by means of an Integrated Safety Analysis (ISA). The purpose of the ISA is to ensure relevant hazards have been evaluated, and preventive or mitigative features included in the repository design, such that the limits on radiation exposures specified in proposed 10 CFR Part 63 will not be exceeded. The ISA provides a framework for risk-informed, performance-based decision making that is applied to identification of SSCs important to safety, measures for providing defense-in-depth, license specifications and surveillance intervals. Preliminary analyses based on the conceptual design show large margins between expected performance and the regulatory limits.

At present, the major issue identified for preclosure operations is reaching a consensus on the design bases for handling of the waste containers (i.e., the transport casks, disposable canisters, and waste packages) during performance of activities at the repository. This revision of the Strategy is based on a design basis for container drop events that requires either a) a demonstration that releases from a credible design basis drop and breach result in radiation doses below the regulatory limit; or b) a demonstration that the containers are capable of surviving a credible design basis drop event without a breach of the containment boundary. The present strategy for Navy and commercial spent fuel disposable canisters is to demonstrate that releases from credible design basis drop and breach events result in radiation doses below the regulatory limit. Analysis based on the conceptual design indicate that the consequences of a breach of the disposable canisters containing Navy spent fuel are acceptable, i.e., the resulting offsite dose is below the regulatory limit. The present strategy for disposable containers for DOE spent fuel and for waste packages is to demonstrate that the containers are capable of surviving a credible design basis drop event without a breach of the containment boundary. This issue will be resolved prior to submittal of the License Application.

CONTENTS

Page

EXECUTIVE SUMMARY – PRECLOSURE SAFETY STRATEGY v

ACRONYMS xiii

1. INTRODUCTION 1-1
   1.1 OBJECTIVE AND SCOPE 1-1
   1.2 APPLICABILITY OF QA CONTROLS 1-1
   1.3 OVERVIEW OF PRECLOSURE SAFETY STRATEGY 1-1

2. REQUIREMENTS 2-1
   2.1 PERFORMANCE OBJECTIVES FOR THE GEOLOGIC REPOSITORY OPERATIONS AREA THROUGH PERMANENT CLOSURE 2-2
   2.2 INTEGRATED SAFETY ANALYSIS 2-3

3. ELEMENTS OF THE PRECLOSURE SAFETY CASE 3-1

4. INTEGRATED SAFETY ANALYSIS APPROACH 4-1
   4.1 HAZARDS ANALYSIS AND SELECTION OF DESIGN BASIS EVENTS 4-1
      4.1.1 Event Identification 4-1
      4.1.2 Event Categorization 4-3
      4.1.3 Event Consequence Analysis Methods 4-4
   4.2 DESCRIPTION OF POTENTIAL HAZARDS, EVENT SEQUENCES, AND CONSEQUENCES 4-6
      4.2.1 Internal Events with Potential Releases 4-6
      4.2.2 Internal Events with No Release 4-8
      4.2.3 Beyond-Design-Basis Events 4-10

5. USE OF DESIGN MARGIN AND DEFENSE-IN-DEPTH IN PRECLOSURE REPOSITORY SYSTEM DESIGN AND OPERATIONS 5-1
   5.1 INTRODUCTION 5-1
   5.2 DEFINITION OF DESIGN MARGIN 5-1
   5.3 DEFINITION OF DEFENSE-IN-DEPTH 5-2
   5.4 RELATIONSHIP OF MARGIN AND DEFENSE-IN-DEPTH 5-3
   5.5 TRADITIONAL REACTOR ENGINEERING PRACTICE 5-3
   5.6 RISK-INFORMED DESIGN PROCESS 5-4
   5.7 REPOSITORY SAFETY CASE 5-5

6. STRATEGY FOR PREVENTION OR MITIGATION OF PRECLOSURE OFFSITE RADIATION EXPOSURE 6-1
   6.1 IDENTIFICATION OF IMPORTANT TO RADIOLOGICAL SAFETY FEATURES AND CONTROLS 6-1
   6.2 DESIGN BASES FOR FACILITIES AND LIMITS ON OPERATIONS 6-2
   6.3 SAFETY STRATEGY FOR REPOSITORY PRECLOSURE OPERATIONS FUNCTIONS 6-2

7. DESIGN BASIS EVENT CONSEQUENCE ANALYSIS 7-1

8. QUALITY CLASSIFICATION ANALYSIS 8-1
8.1 QUALITY LEVEL 1 (QL-1) 8-1
8.2 QUALITY LEVEL 2 (QL-2) 8-2
8.3 QUALITY LEVEL 3 (QL-3) 8-2
8.4 CONVENTIONAL QUALITY (CQ) 8-3

9. REFERENCES 9-1
9.1 DOCUMENTS CITED 9-1
9.2 CODES, STANDARDS, AND REGULATIONS 9-2

APPENDIX A – PRECLOSURE SAFETY STRATEGY GLOSSARY A-1

FIGURES

Page

3-1. Elements of the Preclosure Safety Case 3-1
4-1. Integrated Safety Analysis 4-2
4-2. Event Identification 4-3
5-1. Application of Design Margin and Defense-in-Depth in a Risk-Informed Design Process 5-5
8-1. Quality Level Classification Criteria 8-4

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TABLES

Page

Table 4-1. Radiation Dose Limits for Category 1 and Category 2 Events 4-5
Table 4-2. Category 1 Design Basis Events 4-7
Table 4-3. Category 2 Design Basis Events 4-7
Table 4-4. Internal Events with No Release 4-8
Table 4-5. Beyond-Design-Basis Events 4-11
Table 5-1. Defense-in-Depth for Nuclear Power Plants 5-2
Table 5-2. Defense-in-Depth for the Repository Preclosure Safety Case 5-3
Table 6-1. Repository Preclosure Safety Strategy 6-3
Table 7-1. Preclosure Dose Calculation Results for SRCR Reference Design 7-3

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ACRONYMS

ALARA	As Low As Is Reasonably Achievable
ATS	Assembly Transfer System
CFR	Code of Federal Regulations
CPB	Carrier Preparation Building
CTS	Canister Transfer System
CQ	Conventional Quality
DBE	Design Basis Event
DC	Disposal Container
DCHS	Disposal Container Handling System
DOE	U.S. Department of Energy
EPA	Environmental Protection Agency
GROA	Geologic Repository Operations Area
HLW	High Level Waste
HEPA	High-efficiency Particulate Air (filter)
HVAC	Heating, ventilation and air conditioning
ISA	Integrated Safety Analysis
LA	License Application for a construction authorization for the Yucca Mountain Repository
M&O	Management and Operating Contractor
MGR	Monitored Geological Repository
NUREG	NRC regulatory guidance document
NRC	U.S. Nuclear Regulatory Commission
QA	Quality Assurance
QL	Quality Level
RSS	Repository Safety Strategy
RSS4	Repository Safety Strategy Revision 4
SCP	Site Characterization Plan
SFA	Spent Fuel Assembly
SNF	Spent Nuclear Fuel
SR	Site Recommendation
SRCR	Site Recommendation Consideration Report
SRP	NRC Standard Review Plan
SSC	Structure, system or component

ACRONYMS (Continued)

TEDE	Total Effective Dose Equivalent
WP	Waste Package
WHB	Waste Handling Building
WTB	Waste Transfer Building

1. INTRODUCTION

1.1 OBJECTIVE AND SCOPE

A key issue for the site recommendation and license application considerations for a high-level waste repository at Yucca Mountain is the ability of that site to protect the public from any undue radiological risk before and after permanent closure. The U.S. Department of Energy (DOE) is therefore preparing a case regarding preclosure and postclosure safety of a potential repository system at Yucca Mountain. This document, Revision 4 of the Repository Safety Strategy (RSS), reports the current status of this safety case and the plans to complete it.

This document updates its predecessors to reflect project developments and information learned since previous revisions. This revision is based on the work to date, including RSS workshops conducted from February through June 2000. While previous revisions focused solely on postclosure safety, Revision 4 also addresses the preclosure safety case. This volume, Volume I, presents the preclosure safety strategy. Volume II presents the postclosure safety strategy.

1.2 APPLICABILITY OF QA CONTROLS

Revision 4 of the RSS is subject to controls specified in the *Quality Assurance Requirements and Description* (DOE 2000). Revision 4 has been planned, developed, checked, reviewed, and approved in accordance with applicable quality assurance procedures.

This revision of the RSS may be affected by technical product input information that requires confirmation. Any changes to the document that may occur as a result of completing the confirmation activities will be reflected in subsequent revisions. The status of the input information quality may be confirmed by review of the Document Input Reference System database.

1.3 OVERVIEW OF PRECLOSURE SAFETY STRATEGY

In this revision of the Repository Safety Strategy, each of the elements of the repository preclosure safety case is explained, and the safety strategy for the design and operation of the surface and underground operational facilities is presented.

The preclosure safety strategy focuses on the regulatory performance objectives for the repository system through permanent closure. These performance objectives will eventually be codified in the regulation of the Nuclear Regulatory Commission (NRC) that applies to the repository system. This version of the strategy focuses on the performance objectives included in NRC's proposed regulation, 10 CFR Part 63 (64 FR 8640). The following elements of the repository preclosure safety case combine to provide confidence these performance objectives would be met:

- Integrated safety analysis (ISA)
- Margin and defense-in-depth
- Consequence analysis of beyond-design-basis events
- Commercial Nuclear Industry precedent and experience
- License specifications and surveillances

Each of these five elements is consistent with historic and evolving approaches taken by the Nuclear Regulatory Commission in evaluating whether nuclear facility development and operation will be conducted in a manner that will adequately protect public health and safety.

Proposed 10 CFR Part 63.111 (64 FR 8640) specifies that compliance with the NRC's repository preclosure performance objectives for the Yucca Mountain repository will be demonstrated through the ISA. The ISA's purpose is to ensure relevant hazards that could result in unacceptable consequences have been evaluated, and preventive or mitigative features included in the repository design, such that the limits on radiation exposures specified in proposed 10 CFR Part 63.111 (64 FR 8640), will not be exceeded. The ISA provides the risk insights essential to the risk-informed, performance-based decision making that is applied to identification of structures, systems, and components (SSCs) important to safety, measures for providing defense-in-depth, license specifications and surveillance intervals. The ISA identifies the potential natural and operational hazards for the preclosure period; assesses potential design basis event (DBE) sequences and their consequences; and identifies the SSCs and activities of personnel intended to prevent or mitigate each accident sequence. DBE identification and analysis is an iterative process integral to repository design. Consequently, the preclosure safety analysis and DBEs will continue to evolve as the design matures.

The ISA provides for an allocation of event prevention, or consequence mitigation, to the design of the surface and subsurface systems, structures, components, and equipment. It provides explicit consideration of concurrent construction and waste-handling operations based on the proposed phased construction sequence and the operational functions to be performed during each phase of repository development and operation.

The ISA approach to external events and natural phenomena is based on successful NRC commercial power reactor and other nuclear fuel-cycle facility licensing precedent and is generally consistent with the applicable portions of the approach contained in the NRC's draft NUREG 1513, Integrated Safety Analysis Guidance Document (Milstein 2000). Radionuclide releases from the geologic repository operations area (GROA) will be prevented by designing the repository SSCs important to safety to withstand the bounding criteria associated with credible external events.

The preclosure safety strategy is to design, construct, and operate SSCs that are important to safety to survive credible internal and external events and natural phenomena in a manner that the design basis event dose limits are not exceeded.

The safety strategy for internal events will be based on the repository preclosure operations functions of: (1) waste receipt; (2) transfer of waste to disposal containers (3) disposal container closure welding; (4) transfer to the underground drifts; (5) emplacement; and (6) monitoring until repository closure. The safety strategy for each of these basic functions will be to design and operate SSCs important to safety so as to prevent adverse events and conditions if possible while at the same time applying mitigation measures if required to ensure compliance with the dose limits.

Proposed 10 CFR Part 63 (64 FR 8640) requires that measures to control criticality be included in the ISA. Criticality control was not addressed in-depth during the development of this initial

safety case strategy but will be evaluated during development of the final safety case for the license application to demonstrate that the design criteria and proposed operating procedures for SSCs important to safety are adequate to prevent all credible criticality events.

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2. REQUIREMENTS

The NRC specifies performance objectives in proposed 10 CFR Part 63.111 (64 FR 8640) to ensure the geologic repository operations area (GROA) is designed and operated to protect against radiation exposures and releases of radioactivity prior to permanent closure. Specifically, for protection of the workers and general public 10 CFR Part 63.111(a) requires (1) the exposure limits codified at 10 CFR Part 20 to be met and (2) during normal operations and anticipated operational occurrences, the annual dose to any real member of the public, located beyond the boundary of the site, to be limited to a total effective dose equivalent (TEDE) of 0.25 mSv (25 mrem). Additionally, at 10 CFR Part 63.111(b) numerical guides for design objectives are specified for design basis events (DBEs). The Energy Policy Act of 1992 directs the Environmental Protection Agency (EPA) to develop public health and safety standards for radioactive material stored or disposed of in the potential repository at Yucca Mountain and for the NRC to conform its final regulations to these standards. The EPA has proposed standards specific to Yucca Mountain to be included in a new 40 Part 197 to the Code of Federal Regulations (64 FR 46976).

DBEs are a combination of postulated challenges and failure events against which facilities are designed to ensure adequate and safe response, i.e., capability to prevent or mitigate the consequences of accidents and severe natural phenomena. The proposed regulation distinguishes between Category 1 and Category 2 DBEs. Category 1 DBEs are those events expected to occur one or more times before permanent closure. Included in Category 1 events are those that occur regularly or moderately frequently, and are sometimes identified as "normal operational activities" associated with receiving, handling, packaging, storing, emplacing, and retrieving high-level waste. Also included in Category 1 are "anticipated operational occurrences," including accidents, postulated to occur one or more times during the operating lifetime of the facility. Category 2 DBEs are those events that have low probability of occurrence, but have at least one chance in 10,000 of occurring before permanent closure (64 FR 8640). Assuming a preclosure operational period of 100 years, for event sequences whose probability is uniform throughout the period, this corresponds to an average annual probability of occurrence of one chance in 1 million. Category 2 events are unlikely but sufficiently credible to warrant consideration. Events with a probability of less than one chance in 10,000 years are not considered credible during the preclosure period.

The NRC is proposing (64 FR 8640) that compliance with the preclosure performance objectives would be demonstrated through an ISA of the GROA. The ISA is a systematic examination of potential hazards at the GROA. It identifies the potential natural and operational hazards, the potential for accident event sequences, and describes potential event sequences and their consequences, as well as the SSCs, equipment, and actions of personnel relied upon to prevent or mitigate the accident sequence. Its purpose is to ensure that relevant hazards that could result in unacceptable consequences have been adequately evaluated and appropriate protective measures have been identified such that the GROA will comply with the preclosure requirements for protection against radiation exposures specified in proposed 10 CFR Part 63.111 (64 FR 8640). In addition the ISA is also intended to ensure that possible interactions of events or operations do not combine to increase risks. As used in proposed 10 CFR Part 63, integrated means combined consideration of safety measures that, considered separately, might not achieve the overall health and safety protection desired. Such integration would include, but not be limited to, integration of

fire protection, radiation safety, criticality safety, and chemical safety measures. A fundamental aspect of the ISA is the identification and analysis of Category 1 and Category 2 DBEs. Category 1 events as described above include "normal operations" and "anticipated events," while Category 2 events are those unlikely but credible events that could challenge the design of the GROA to maintain radiation exposures within allowable limits. The analysis of a specific DBE would include identification of an event sequence comprising an initiating event (e.g., an earthquake) and the associated combinations of repository system events and component failures that can potentially lead to exposure of individuals to radiation. An example DBE is a postulated earthquake (the initiating event) that could result in (1) failure of a crane lifting a spent fuel waste package inside a waste handling building, (2) damage to the building ventilation (filtration) system, (3) drop and breach of the waste package, (4) damage to the spent fuel, (5) release of some radioactive material through the damaged ventilation (filtration) system, and (6) exposure of an individual (either a worker or a member of the public) to the released radioactive material.

2.1 PERFORMANCE OBJECTIVES FOR THE GEOLOGIC REPOSITORY OPERATIONS AREA THROUGH PERMANENT CLOSURE

Proposed 10 CFR Part 63.111(a)(2) (64 FR 8640) specifies the following radiation dose limits to the public from normal operations and Category 1 events:

> During normal operations, and for Category 1 design basis events, the annual dose to any real member of the public, located beyond the boundary of the site shall not exceed a TEDE of 0.25 mSv (25 mrem).

Proposed 10 CFR Part 63 (64 FR 8640) specifies radiation dose limits for workers during normal operation and, from Category 1 events, by reference to 10 CFR Part 20.1201(a) (10 CFR 20):

> (1) An annual limit, which is the more limiting of --
>
> (i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or
>
> (ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).
>
> (2) The annual limits to the lens of the eye, to the skin, and to the extremities, which are:
>
> (i) A lens dose equivalent of 15 rems (0.15 Sv), and
>
> (ii) A shallow-dose equivalent of 50 rems (0.50 Sv) to the skin or to any extremity."

Proposed 10 CFR Part 63.111 (b)(2) (64 FR 8640) specifies radiation dose limits for members of the public from Category 2 events as follows:

> The geologic repository operations area shall be designed so that taking into consideration Category 2 design basis events and until permanent closure has been completed, no individual located on, or beyond, any point on the boundary of the

site, will receive the more limiting of a TEDE of 0.05 Sv (5 rem), or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent shall not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin shall not exceed 0.5 Sv (50 rem).

There are no radiation dose limits for workers from Category 2 events.

2.2 INTEGRATED SAFETY ANALYSIS

Proposed 10 CFR Part 63.111(c) (64 FR 8640) specifies that:

> An integrated safety analysis of the geologic repository operations area that meets the requirements specified at Sec. 63.112 shall be performed. This analysis shall include a demonstration that:
>
> 1) The requirements of Sec. 63.111(a) will be met; and
> 2) The design meets the requirements of Sec. 63.111(b).

Proposed 10 CFR Part 63.112 (64 FR 8640) specifies the requirements for integrated safety analysis of the geologic repository operations area.

> The integrated safety analysis of the geologic repository operations area shall include:
>
> (a) A general description of the structures, systems, components, equipment, and process activities at the geologic repository operations area.
>
> (b) An identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area, including a comprehensive identification of potential accident/event sequences that would result in unacceptable consequences (i.e., design basis events).
>
> (c) Data pertaining to the Yucca Mountain site, and the surrounding region to the extent necessary, used to identify naturally occurring and human-induced hazards at the geologic repository operations area.
>
> (d) The technical basis for either inclusion or exclusion of specific, naturally occurring and human-induced hazards in the safety analysis.

(e) An analysis of the performance of the major design structures, systems, and components, both surface and subsurface, to identify those that are important to safety, including identification and description of controls that are relied on to limit or prevent potential accidents or mitigate their consequences, and including identification of measures taken to ensure the availability of identified safety systems. The analysis required in this paragraph shall include, but not necessarily be limited to, consideration of:

(1) Means to limit concentration of radioactive material in air;

(2) Means to limit the time required to perform work in the vicinity of radioactive materials;

(3) Suitable shielding;

(4) Means to monitor and control the dispersal of radioactive contamination;

(5) Means to control access to high radiation areas or airborne radioactivity area;

(6) Means to control criticality;

(7) Radiation alarm system to warn of significant increases of radiation levels, concentrations of radioactive material in air, and increased radioactivity in effluents;

(8) Ability of systems, structures, and componentsstructures, systems, and components to perform their intended safety functions, assuming the occurrence of design basis events;

(9) Explosion and fire detection systems and appropriate suppression systems;

(10) Means to control radioactive waste and radioactive effluents, and permit prompt termination of operations and evacuation of personnel during an emergency;

(11) Means to provide reliable and timely emergency power to instruments, utility service systems, and operating systems important to safety if there is a loss of primary electric power;

(12) Means to provide redundant systems necessary to maintain, with adequate capacity, the ability of utility services important to safety; and

(13) Means to inspect, test, and maintain structures, systems, and components important to safety, as necessary, to ensure their continued functioning and readiness.

(f) A description and discussion of the design, both surface and subsurface, of the geologic repository operations area, including:

(1) The relationship between principal design criteria and the requirements specified at Sec. 63.111(a) and (b); and

(2) The design bases and their relation to the principal design criteria.

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3. ELEMENTS OF THE PRECLOSURE SAFETY CASE

In order to enhance confidence that the repository preclosure performance objective will be met, the DOE considers it prudent to provide more than the engineered and administrative controls identified by an integrated safety analysis in its preclosure safety case. The preclosure safety case therefore includes the five elements as listed in Figure 3-1.

Figure 3-1. Elements of the Preclosure Safety Case

- ✓ Integrated Safety Analysis
- ✓ Margin and Defense-in-depth
- ✓ Consequence Analysis of Beyond-design-basis Events
- ✓ Nuclear Industry Precedent and Experience
- ✓ License Specifications and Surveillances

NRC's proposed regulation (64 FR 8640) acknowledges the value of a comprehensive safety case and emphasizes that the standard for compliance with the preclosure performance objective is reasonable assurance. In (NRC 1999) the NRC states that the risk-informed, performance-based regulatory philosophy underlying this proposed regulation rests on a foundation of safety margin and defense-in-depth. The preclosure safety case therefore provides these and other elements to support NRC's ability to find, with reasonable assurance, that the preclosure performance objective will be met. Each of the five elements is described in the following text.

Integrated Safety Analysis–The ISA provides the quantitative estimate of repository preclosure performance. This includes development of the technical basis required to satisfy the requirement of NRC's proposed rule in 10 CFR Part 63.111 (64 FR 8640). Section 4 discusses the ISA in detail.

Margin and Defense-in-depth–Use of margin is a standard engineering practice of including capability in excess of minimum design requirements in specifications for engineered SSCs and a standard operational practice of establishing conservative operating limits with respect to the minimum design requirements. Margin provides for a reserve capability in order to account for uncertainty in the conditions to which the SSCs will be subjected, to account for variability in the properties of component materials, and to account for the possibilities of overload or understrength due to variations in material dimensions, construction procedures, or simplifications of calculations. Defense-in-depth is the term used to describe the decision to include in the design and operation of a facility multiple diverse and redundant physical and administrative barriers or other protective measures to mitigate unanticipated conditions, processes, and events such that failure of any one barrier does not result in failure of the entire system. Margin and defense-in-depth are discussed in detail in Section 5.

Consequence Analysis of Beyond-design-basis Events–The preclosure design basis for the repository precludes a breach of a high level waste (HLW) or spent fuel shipping cask, disposable canister or waste package due to a drop event or other credible impacts unless it can be demonstrated that releases from a breached container result in site boundary doses below the regulatory limits without taking credit for the high-efficiency particulate air (HEPA) filters in the ventilation system. The consequences of a breach of the disposable canisters containing Navy spent fuel are acceptable, i.e., the resulting offsite dose is below the regulatory limit. The consequences of a breach of a DOE or commercial spent fuel canister have not been analyzed using fully qualified data. In order to provide confidence in the repository preclosure design, the safety strategy includes analysis of the consequences of these events, using the best estimate of expected conditions, to demonstrate that even for these highly improbable, beyond-design-basis events, offsite doses would meet NRC's regulatory criteria.

Nuclear Industry Precedent and Experience–Commercial nuclear industry and other nuclear fuel cycle facility precedent and experience will be used, where appropriate, in the design and analysis of the repository operational facilities to reduce or eliminate uncertainty in the conditions to which the SSCs will be subjected, to account for variability in the properties of component materials, and to account for the possibilities of overload or understrength due to variations in material dimensions, construction procedures, or simplifications of calculations. Commercial nuclear industry and other nuclear fuel cycle facility precedent also provides confidence in the ISA approach and provides data and lessons learned for direct incorporation into the hazards and consequence analyses.

License Specifications and Surveillances–License specifications are the rules that establish when repository SSCs that have been determined to be important to safety by the ISA must be operable (including allowed outage times) and establish limiting parameters for the operation of systems important to safety and limits on the types and form of waste to be received. This is to provide additional assurance that the repository preclosure operations functions will be performed safely. Surveillances are the periodic (monthly, quarterly, annually, etc.) operational tests of the various SSCs to demonstrate the ability to perform their required safety functions. Proposed 10 CFR Part 63.21(13) (64 FR 8640) requires that license specifications, including the identification and justification for the selection of those variables, conditions, or other items that are determined to be probable subjects of license specifications, be included in the repository license application. The work necessary to develop the proposed license specifications and surveillance tests has not yet been initiated. This work must be completed to support a license application for the repository.

4. INTEGRATED SAFETY ANALYSIS APPROACH

The ISA provides the risk insights essential to risk-informed, performance-based decision making that is applied to identification of SSCs important to safety, measures for providing defense-in-depth, license specifications and surveillance intervals. As stated in Section XIII of the Supplementary Information accompanying the NRC's proposed 10 CFR Part 63 (64 FR 8640),

> The information the Commission needs to make a finding of reasonable assurance that the GROA will comply with the risk-informed, preclosure requirements at 10 CFR Part 63.111, will be provided by the ISA. The Commission proposes criteria, at 10 CFR Part 63.112, for the content of the ISA.

The ISA employs qualitative analyses as well as quantitative analytic probabilistic risk assessment techniques, such as event tree, fault tree, and uncertainty analyses. The ISA methodology is consistent with the guidance contained in NRC's draft NUREG 1513, Integrated Safety Analysis Guidance Document (Milstein 2000). The elements of the ISA are illustrated in Figure 4-1. These elements include systematic review of the site and facility features and processes to define the type of events that can occur. Event sequences identified include the full range of probable events, from normal operational events anticipated to occur routinely to very low-probability events. Event sequences are identified by detailed site- and design-specific event scenario development, where event sequences are defined and the anticipated frequency of occurrence of events is established. Figure 4-1 also shows the role of consequence analysis and radiological dose assessment in the ISA. The ISA accommodates the identification and evaluation of uncertainties in models and parameters in the frequency and consequence analyses. The uncertainty analyses provide the bases for identifying measures for defense-in-depth.

The ISA performs an important role in the design process. Based on insights and results obtained from the safety assessment, the acceptability of the design can be established. Alternative design approaches or additional design features are identified based on preliminary safety assessment results, as part of the iterative design process. The ISA also plays a key role in the identification of design requirements, facility features, and operational and maintenance controls important to safety and is a primary input to the quality assurance classification process.

4.1 HAZARDS ANALYSIS AND SELECTION OF DESIGN BASIS EVENTS

4.1.1 Event Identification

As illustrated in Figure 4-2, events that may result in radiological consequences to the public and facility workers during the repository preclosure period are identified based on a comprehensive review of repository site characteristics, facility design features, and operational processes to be performed. As shown in Figure 4-2, a thorough analysis of the internal and external hazards associated with preclosure operations of the repository is performed. Internal hazards are those hazards presented by operation of the facility and associated processes. The types of internal hazards considered include collisions, crushing, fire, explosions, implosions, chemical contamination, flooding, radiation and radioactive material, magnetic, thermal, electrical, and fissile material. External hazards involve natural phenomena, such as earthquakes and volcanism,

and external man-made hazards, such as those posed by aircraft and nearby military/industrial facilities.

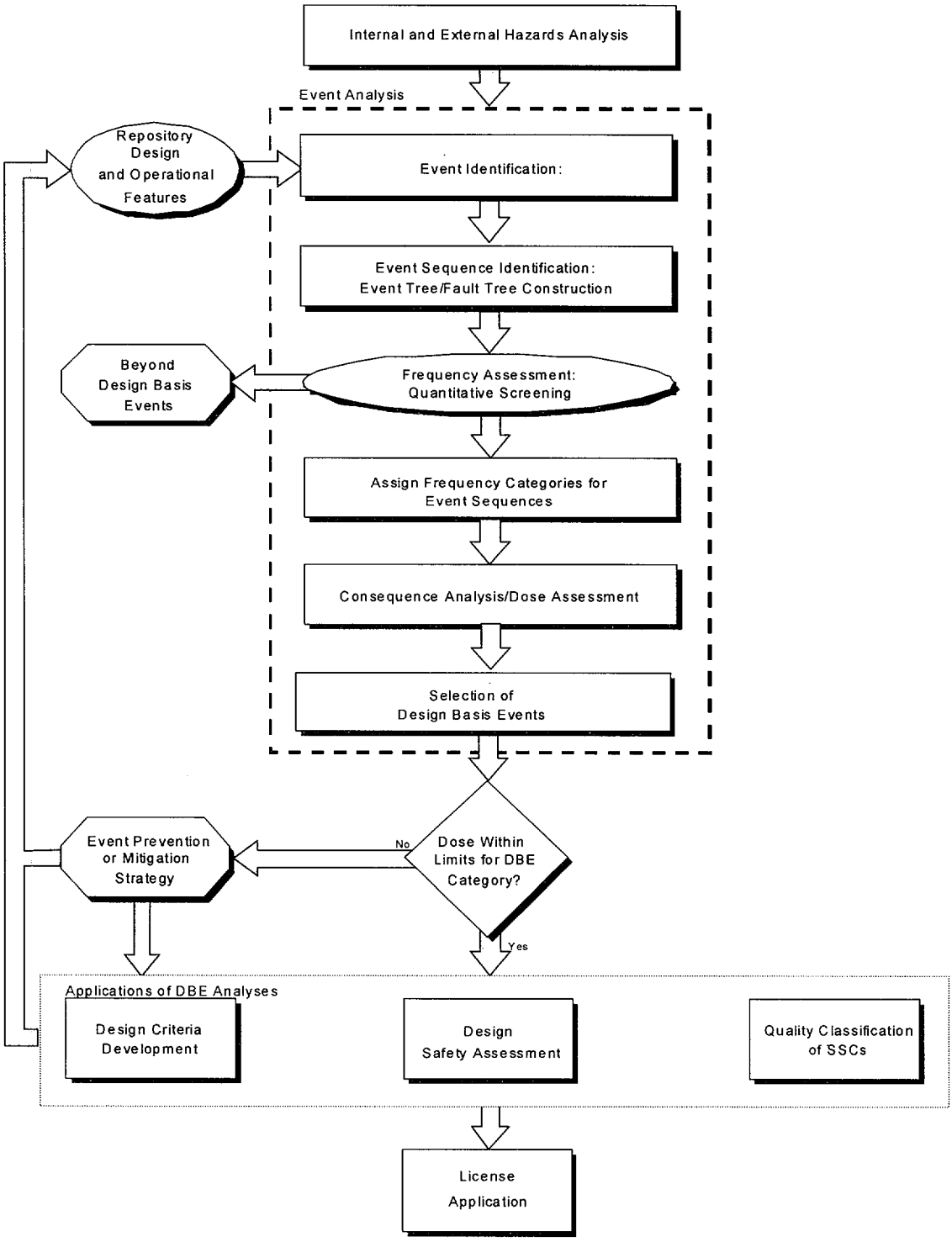

Figure 4-1. Integrated Safety Analysis

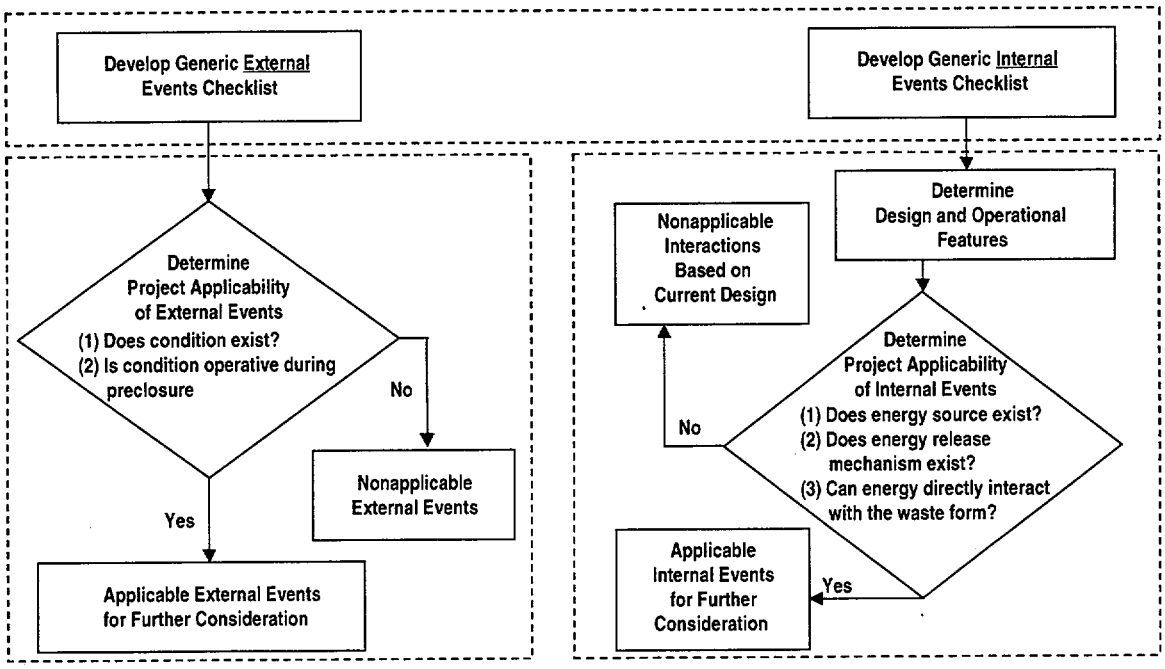

Figure 4-2. Event Identification

Once the repository site characteristics and the preliminary facility design and operational processes are defined, they are evaluated against specified criteria to determine the potential for events that could result in radiological consequences. The criteria used to determine the applicability of internal and external events to repository facilities and processes are listed in Sections 5.1.1.3.1 and 5.1.1.3.2, respectively, of the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a).

This evaluation results in a list of initiating events to be considered in the design and safety analysis processes. The list of applicable events is used as an input to the scenario development process, where event sequences are defined and event frequencies of occurrence are estimated based on a detailed, design-specific evaluation.

4.1.2 Event Categorization

Event sequences begin with an initiating event (such as an earthquake), are generally followed by one or more additional events, and sometimes conclude with a radiological release from the GROA. Event sequences are developed using event trees. Event trees are diagrams that depict the chronological sequence of events that could result from an initiating event. The product of an event tree analysis is a set of event sequences. The frequency of occurrence for each event sequence is determined by multiplying the initiating event frequency by the conditional probability of each successive event in the event sequence. For internal events, initiating event frequencies are typically determined by multiplying the failure rate of an initiating event (e.g., fractional number of drops per lift) by the maximum number of handlings in a single year (e.g., number of lifts per year). Equipment failure rates and conditional event failure probabilities (e.g., HVAC system failure) are determined using fault tree analysis or historical equipment data.

The result of the event sequence development process is a list of event sequences with a corresponding quantitative frequency of occurrence. Event sequences are categorized based on frequency of occurrence by the NRC in proposed 10 CFR Part 63.2 (64 FR 8640). First, event sequences are categorized as DBEs or beyond-design-basis events. DBEs are further subdivided into two types: Category 1 and Category 2. Category 1 DBEs are those natural and human-induced event sequences expected to occur one or more times before permanent closure. This is equivalent to an average annual frequency of at least one in one hundred (0.01 per year) based on a 100-year repository preclosure operational period. Category 2 DBEs are other natural and human-induced event sequences that have at least one chance in 10,000 of occurring before permanent closure. This is equivalent to an average annual frequency of one in one million (0.000001 per year) based on a 100-year preclosure operational period. Event sequences that have less than one chance in 10,000 of occurring before permanent closure of the repository are known as beyond-design-basis events. Proposed 10 CFR Part 63 does not require evaluation of the consequences of beyond-design-basis events.

Separate consequence acceptance criteria are established for Category 1 and Category 2 DBEs. These consequences are usually expressed in terms of radiation doses[1] to workers or members of the public. The radiation doses to members of the public are required to be less than the limits in proposed 10 CFR Part 63—a 25 mrem/yr limit[2,3] for Category 1 design basis events (10 CFR Part 63.111(a)(2)) and a 5 rem per event sequence limit[4,5] for Category 2 design basis events (10 CFR Part 63.111(b)(2)) (64 FR 8640). To implement as low as is reasonably achievable (ALARA) requirements, 10 CFR Part 20.1101(d) (10 CFR Part 20) establishes a 10 mrem/yr constraint on air emissions of radioactive material to the environment. For Category 1 DBEs (including normal operations), radiation doses for repository workers must not exceed the 5 rem/yr limit[6] contained in 10 CFR Part 20.1201(a)(1). NRC regulations (10 CFR Part 20.1101(b)) also require that radiation doses for workers be ALARA.

4.1.3 Event Consequence Analysis Methods

Radiation doses for Category 1 and Category 2 events are calculated using two models: one for modeling chronic exposures resulting from routine operations and Category 1 events, and the other for modeling acute exposures from Category 2 events. The term "chronic" is used to describe the annual exposure to an onsite worker and an individual living at the site boundary and continuously exposed to an average level of exposure. By comparison, "acute" refers to the

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[1] The general term "radiation dose" is used to mean the total effective dose equivalent as used by the NRC in 10 CFR Part 20.1003 (10 CFR Part 20) and proposed 10 CFR Part 63.2 (64 FR 8640), and the annual committed effective dose equivalent as used by the U.S. Environmental Protection Agency in proposed 40 CFR Part 197.2 (64 FR 46976).

[2] 10 CFR Part 20.1301(a)(1) also contains a 100 mrem/yr limit for members of the public and 10 CFR Part 20.1301(a)(2) also contains a 2 mrem/hr limit from external sources of radiation in unrestricted areas.

[3] Proposed 40 CFR Part 197.4(a) also contains a 15 mrem/yr limit

[4] Proposed 10 CFR Part 63.111(b)(2) (64 FR 8640) also contains a 50 rem per event sequence limit for any individual organ or tissue other than the lens of the eye, a 15 rem per event sequence limit for the lens of the eye, and a 50 rem per event sequence limit for the skin.

[5] Proposed 40 CFR Part 197.4(a) (64 FR 46976) does not contain a limit for Category 2 design basis events.

[6] 10 CFR Part 20.1201(a)(1) also contains a 50 rem/yr limit for any individual organ or tissue other than the lens of the eye. 10 CFR Part 20.1201(a)(2) also contains a 15 rem/yr limit for the lens of the eye and a 50 rem/yr limit for the skin.

maximum exposure that an individual at the site boundary is expected to receive during the course of a DBE over a relatively short period, on the order of 2 hours.

Category 1 Events–The total Category 1 annual radiation dose is based on contributions from three sources: Category 1 normal operational (routine) releases from the Waste Handling Building, normal operational (routine) releases from the subsurface areas of the repository, and releases from the GROA from nonroutine events anticipated to occur at an average annual frequency of 0.01 events per year or higher. The releases and exposures from all identified Category 1 design basis events are aggregated and converted to annual dose.

Table 4-1. Radiation Dose Limits for Category 1 and Category 2 Events

	Dose Limit Category 1 Events	Dose Limit Category 2 Events
Dose to Public	25 mrem/yr	5 rem per event sequence
Dose to Workers	ALARA but not greater than 5 rem/yr	

Radiation dose calculations for Category 1 events consider the potential exposure pathways, including inhalation, ingestion, submersion in air, and direct radiation from surface contamination. When analyzing radiation doses from Category 1 events, calculation input parameters, such as atmospheric dispersion factors, breathing rates, and ingestion rates, are based on long-term average data. Radiological source terms for Category 1 radiation dose calculations are also based on average waste characteristics. This practice is consistent with the long-term, or chronic, nature of the individual exposure.

Category 2 Events–In contrast to Category 1 event sequences, Category 2 design basis event sequences are unlikely (i.e., they are not expected to occur in the period before permanent closure.) It is even more unlikely that two or more Category 2 events will occur in any given year. Therefore, it is appropriate to assess the consequences of each potential Category 2 sequence on a per event basis. Only the inhalation and air submersion pathways are considered for Category 2 events. The potential radiation doses from ingestion and contaminated soil are not considered for acute exposures associated with events that are not expected to occur at the repository during the preclosure period because the area would be evacuated and access restricted until contamination was cleaned up. When analyzing Category 2 design basis events, conservative short-term values for input parameters, such as atmospheric dispersion factors or breathing rates, are used consistent with the short-term, or acute, nature of the exposure. For example, meteorological dispersion factors based on short-term conditions are employed in Category 2 radiation dose calculations. Source terms for Category 2 events are also estimated based on conservative fuel characteristics being involved.

4.2 DESCRIPTION OF POTENTIAL HAZARDS, EVENT SEQUENCES, AND CONSEQUENCES

The *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a, Section 5.1) identified the internal and external hazards associated with preclosure operations at the repository.

The *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a, Section 5.2) further analyzed external and internal event sequences and assigned frequency categories to the sequences. Six categories of external events and natural phenomena were found to be sufficiently credible to merit consideration and to have the potential to result in a radiological release from the GROA: (1) loss of offsite power, (2) earthquakes with vibratory ground motion, (3) earthquakes with fault displacement, (4) floods, (5) impacts of tornado-generated missiles, and (6) high winds generated during tornadoes (CRWMS M&O 2000a, Section 5.2.1). The *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a, Section 5.2.1) contains a description of these events.

Internal event sequences were screened into one of the following three groups based on their frequency of occurrence and potential to result in a radiological release from the GROA.

4.2.1 Internal Events with Potential Releases

Event sequences which could result in a release of radioactivity from the GROA will be mitigated by the facility design. The Category 1 DBEs evaluated in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a, Section 5.3.2) are postulated to occur during the handling of bare commercial spent nuclear fuel (CSNF) assemblies or spent nuclear fuel (SNF) assembly baskets in the Assembly Transfer System. Table 4-2 contains a list of these event sequences. This table, which is based on a similar table in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a), contains data that were updated during the development of the Repository Safety Strategy Revision 4. The postulated frequencies of these event sequences range from 0.04 to 0.20 per year.

The Category 2 design basis events evaluated in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a, Section 5.3.3) are postulated to occur as a result of drops or collisions among handling equipment, unsealed disposal containers, or unsealed shipping casks. Table 4-3 contains a list of the Category 2 design basis events. Table 4-3, which is based on a similar table in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a), contains data that were updated during the development of the Repository Safety Strategy Revision 4. The postulated frequencies of these event sequences range from 0.00007 to 0.009 per year.

Table 4-2. Category 1 Design Basis Events

(NOTE 1)

Event Sequence #	Event Description	Location	Frequency (per Year)
1-01	SFA Drop Onto Another SFA in Cask	ATS Pool	0.2
1-02	SFA Collision	ATS Pool	0.04
1-03	SFA Drop Onto Empty Basket	ATS Pool	0.04
1-04	SFA Drop Onto Another SFA in Basket Staging Rack (Lowering Into)	ATS Pool	0.2
1-05	Basket Drop Onto Another Basket in Basket Staging Rack (Lifting Out)	ATS Pool	0.04
1-06	Basket Drop Onto Another Basket in Pool (Transfer Into Pool Storage)	ATS Pool	0.04
1-07	Basket Drop Onto Another Basket in Pool (Transfer Out of Pool Storage)	ATS Pool	0.04
1-08	Basket Drop Onto Transfer Cart or Pool Floor	ATS Pool	0.04
1-09	Basket Drop Back Into Pool	ATS Pool	0.04
1-10	Basket Drop Onto ATS Cell Floor	ATS Cell	0.04
1-11	Basket Drop Onto Another Basket in Dryer	ATS Cell	0.04
1-12	SFA Drop Onto Another SFA in Dryer	ATS Cell	0.2
1-13	SFA Drop Onto ATS Cell Floor	ATS Cell	0.2
1-14	SFA Drop Onto Another SFA in Disposal Container	ATS Cell	0.2

NOTE 1: This table, which is based on a similar table in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a), contains data that were updated during the development of the Repository Safety Strategy Revision 4.

Table 4-3. Category 2 Design Basis Events

(NOTE 1)

Event Sequence #	Event Description	Location	Frequency (per Year)
2-01	SFA Basket Collision During Transfer	ATS Pool	0.007
2-02	Uncontrolled Descent of Incline Transfer Cart	ATS Pool	0.007
2-03	Handling Equipment Drop onto SFA Basket in Pool	ATS Pool	0.002
2-04	Handling Equipment Drop onto SFA Basket in Cell	ATS Cell	0.00007
2-05	Unsealed Disposal Container Collision	Disposal Container Handling Cell	0.002
2-06	Unsealed Disposal Container Drop and Slapdown	Disposal Container Handling Cell	0.008
2-07	Handling Equipment Drop onto Unsealed Disposal Container	Disposal Container Handling Cell	0.0001
2-08	Unsealed Shipping Cask Drop into Cask Preparation Pit	ATS Cask Preparation Pit	0.009
2-09	Unsealed Shipping Cask Drop into Cask Unloading Pool	ATS Pool	0.009

NOTE 1: This table, which is based on a similar table in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a), contains data that were updated during the development of the Repository Safety Strategy Revision 4.

4.2.2 Internal Events with No Release

For some event sequences, features of the design either prevent the event sequence from occurring or prevent a radiological release from the GROA should the event occur. Design features which prevent an event sequence from occurring either physically prevent the event from occurring (e.g., by using passive design features or process controls, such as eliminating, at certain steps, the lifting of transportation casks or canistered waste), or reduce the event sequence frequency below the cutoff frequency of one in one million per year (e.g., by using active design features, such as redundant features in cranes and control systems). Design features that prevent a release are based on the premise that events will occur and that affected SSCs must be designed to prevent the release of radioactivity from the GROA during such an event. Prime examples of this include the waste package design basis events, which establish design bases for the waste package to ensure that the waste package will not breach as a result of credible design basis events. Table 4-4 identifies these event sequences. Table 4-4, which is based on a similar table in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a), contains data that were updated during the development of the Repository Safety Strategy Revision 4.

Table 4-4. Internal Events with No Release

(NOTE 1)

Event Group	Design Basis Event	Event Location	SSCs which Prevent a Release
Shipping Cask-Related	Cask Carrier/Railcar Accident (with impact limiters)	Between Site Boundary and CPB	Shipping Cask
	Sealed Shipping Cask Drop Onto Floor (no impact limiters)	Carrier Bay	Shipping Cask, Carrier/Cask Handling System bridge crane, Lifting Fixtures
	Sealed Shipping Cask Drop Into Cask Preparation Pit (no impact limiters)	Cask Preparation Pit	Shipping Cask, Carrier/Cask Handling System bridge crane, Lifting Fixtures
	Shipping Cask Collision (no impact limiters)	CPB, Carrier Bay, or En-Route Between	Shipping Cask
	Handling Equipment Drops onto Cask (no impact limiters)	CPB or Carrier Bay	Shipping Cask
	Shield Door Closes on Cask	ATS or Canister Transfer System (CTS) Airlock	Shipping Cask
	Cask Cooldown System Overpressurization	ATS Cask Preparation Pit	Cask and Dual Purpose Canister Preparation Cooling System
SFA-Related	Flooding Due to Uncontrolled Pool Water Fill/Draindown	ATS Pool & Surrounding Area	Pool Water Level Control
Canister-Related	Canister Drop (NOTE 2)	CTS Cell	Canisters, CTS bridge crane, Lifting Fixtures
	Handling Equipment Drops onto Canister	CTS Cell	Canister, CTS bridge crane, Lifting Fixtures
Falling Objects Strike Disposal Container or Waste Package	WHB Handling Equipment Falls on Disposal Container or Waste Package	Disposal Container Handling System (DCHS) Cell	Disposal Containers, Waste Package, DCHS bridge cranes, Lifting Fixtures, Waste Package Horizontal Lifting System
	Underground Handling Equipment Falls on Waste Package	Subsurface	Waste Package
	Drift Liner / Ground Support Falls on Waste Package	Subsurface	Waste Package

Table 4-4. Internal Events with No Release (Cont'd)

Event Group	Design Basis Event	Event Location	SSCs Credited to Prevent a Release
Waste Package Drops	Aboveground Lifting System Drops Vertically Oriented Waste Package	DCHS Cell	Waste Package
	Aboveground Lifting System Drops Horizontally Oriented Waste Package	DCHS Cell	Waste Package
	Bed Plate Rolls Out of Waste Package Transporter	Subsurface	Waste Package
	Emplacement Gantry Drops Horizontally Oriented Waste Package	Subsurface	Waste Package
	Waste Package Falls Onto a Sharp Object	DCHS Cell or Subsurface	Waste Package
Waste Package Slapdown	Waste Package Tips Over and Slaps Down on a Flat Surface	DCHS Cell	Waste Package
Waste Package Collisions	Waste Package Collides in Receipt and Inspection Area	DCHS Cell	Waste Package
	Transporter Collisions at Normal Operating Speeds	Subsurface	Waste Package
	Transporter Derails w/o Tip-over, but with Waste Package Restraint Failure	Subsurface	Waste Package
	Transporter Derails with Tip-over	Subsurface	Waste Package
	Transporter Door Closes on Waste Package	Subsurface	Waste Package
	Operation of Emplacement Gantry Causes Waste Package Collision	Subsurface	Waste Package
Missiles and Explosive Overpressure	Pressurized System Missile Strikes Waste Package	DCHS Cell	Waste Package
	Waste Package Missile Strike from Battery Hydrogen Explosion	DCHS Cell	Waste Package
Fire and Thermal Hazards	Thermally Overloaded Waste Package	DCHS Cell	Waste Package
	Fire in Disposal Container Cell	DCHS Cell	Waste Package
	Transporter Breakdown Outside North Portal (Insolation)	Between Waste Handling Building (WHB) and North Portal	Waste Package
	Thermally Overloaded Emplacement Drift	Subsurface Emplacement Drift	Waste Package
	Loss of Underground Ventilation	Subsurface	Waste Package
	Rock Fall Buries Waste Package with Debris	Subsurface Emplacement Drift	Waste Package

Table 4-4. Internal Events with No Release (Cont'd)

Event Group	Design Basis Event	Event Location	SSCs which Prevent a Release
Waste Package Overpressure	Fuel Rod Rupture	DCHS Cell	Waste Package
Waste Package Criticality	Waste Package Criticality Misload Flooded Disposal Container	DCHS Cell	Disposal Containers, Waste Package DCHS Cell

NOTE 1: This table, which is based on a similar table in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a), contains data that were updated during the development of the Repository Safety Strategy Revision 4.

NOTE 2: The design basis for the canisters requires that they be capable of surviving a credible design basis drop event without breaching the containment boundary unless it can be demonstrated that releases from a breached canister result in site boundary doses below the regulatory limits without taking credit for the HEPA filters in the ventilation system. The consequences of a breach of the disposable canisters containing Navy spent fuel are acceptable, i.e., the resulting offsite dose is below the regulatory limit.

4.2.3 Beyond-Design-Basis Events

Beyond-design-basis events are internal event sequences that have less than 1 chance in 10,000 of occurring before permanent closure. Event sequences satisfying this frequency criterion do not need to be considered in the design basis, per proposed 10 CFR Part 63 (64 FR 8640). Assuming a preclosure operational period of 100 years, this corresponds to an annual probability of less than 1 chance in 1,000,000 of occurring per year. Event sequences having an annual probability of less than 1 chance in 1,000,000 per year have been considered beyond the design basis and have not been included in the analyses reported in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a). Analyses of beyond-design-basis events may be used to evaluate the repository preclosure design margin and defense-in-depth. For example, the preclosure design basis for the repository precludes a breach of a HLW or spent fuel shipping cask, DOE spent fuel canister or a waste package due to a drop event or other credible impacts unless it can be demonstrated that releases from a breached container result in site boundary doses below the regulatory limits without taking credit for the HEPA filters in the ventilation system. The consequences of a breach of the disposable canisters containing Navy spent fuel are acceptable, i.e., the resulting offsite dose is below the regulatory limit. The waste packages will be designed to survive the drop event. The consequences of a breach of a DOE spent fuel canister have not been analyzed using fully qualified data. In order to provide confidence in the repository preclosure design, an analysis of the consequences of these events, using the best estimate of expected conditions, will be completed to demonstrate that even for this highly improbable, beyond-design-basis event, offsite doses would meet NRC's regulatory criteria.

Beyond-design-basis events are identified in Table 4-5. Table 4-5, which is based on a similar table in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a), contains data that were updated during the development of the Repository Safety Strategy Revision 4. These beyond-design-basis events are internal event sequences that have been specifically addressed in the design process to ensure the event sequences do not occur (i.e., ensure the expected probability is less than one chance in 10,000 during the preclosure period). The beyond-design-basis event sequences identified in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a) would likely be Category 1 or Category 2 DBEs if the

prevention or mitigation feature included in the repository facilities design failed or was not present. These events are beyond-design-basis events because of the deliberate incorporation of design features, physical barriers, administrative controls, or a combination thereof to ensure the sequence of events necessary to result in a radiological release from the GROA has an expected probability of less than 1 chance in 10,000 during the preclosure period.

Table 4-5. Beyond-Design-Basis Events

(NOTE 1)

Event Group	Event (NOTE 2)	Location	Design/ Mitigation Feature
Fire	Fire in Surface Facilities Resulting in Radiological Release	WHB or WTB	Design layout and administrative controls ensure that a credible fire cannot result in a radiological release
	Fire in Subsurface Facilities Resulting in Radiological Release, No Filtration	Subsurface	Design layout and administrative controls ensure that a credible fire cannot result in a radiological release
Shipping Cask-Related	Drop of Shipping Cask from Beyond its Design Basis Height (w/o impact limiters)(i.e., Two-Block Drop), No Filtration	Carrier Bay	WHB Confinement Area Ventilation System, Control Systems, Carrier/Cask Handling System bridge crane, Lifting Fixtures
	Cask Drop into Cask Preparation Pit, No Filtration	ATS Cask Preparation Pit	WHB Confinement Area Ventilation System, Carrier/Cask Handling System bridge crane, Lifting Fixtures
	Non-Mechanistic Shipping Cask Leak (3)	Carrier Bay	Shipping Cask
	Diesel Fire/Explosion Resulting in Breach of a Shipping Cask, No Filtration	Outside CPB	Shipping Cask; No ignition source present to initiate a fire or explosion capable of breaching a cask
SFA-Related	SFA Basket Drop onto Another SFA Basket, No Filtration	ATS Dryer	WHB Confinement Area Ventilation System, Dry Assembly Handling System, Dry Assembly Transfer Machine
	Catastrophic Pool Failure	ATS Pool	WHB Structure, Pool Water Level Control
	Criticality Event in Pool	ATS Pool	Assembly Transfer Baskets, Basket Staging Racks
	Loss of Pool Water Resulting in Zirconium Alloy Cladding Fire	ATS Pool	None – analysis expected to demonstrate there is insufficient heat output to initiate cladding fire (565°C)
	Cladding Failure in the ATS Dryer	ATS Dryer	None – analysis expected to demonstrate there is insufficient heat output to ignite cladding (565°C) and initiate preclosure cladding failure
	Welding Burnthrough of SNF	DCHS Welding Station	None – analysis expected to demonstrate that a welding error resulting in breach of the SNF cladding is physically impossible
Canister-Related	Impact to Disposable Canister that Exceeds its Design Basis (e.g., Two-Block Drop) (NOTE 3)	CTS Cell	WHB Confinement Area Ventilation System, Control Systems, Canister Transfer System bridge crane, Lifting Fixtures
	Criticality Associated with Small Canister Staging Rack	CTS Cell	Small Canister Staging Racks

Table 4-5. Beyond-design-basis Events (Cont'd)

Event Group	Event (NOTE 2)	Location	Design/ Mitigation Feature
Disposal Container/Waste Package-Related	Impact to Waste Package that Exceeds its Design Basis (e.g., Two-Block Drop)	Disposal Container Cell	WHB Confinement Area Ventilation System, Control Systems, DCHS bridge cranes, Waste Package Horizontal Lifting System, Lifting Fixtures
	Unsealed Disposal Container Drop (from normal handling height) onto Cell Floor, No Filtration	Disposal Container Cell	WHB Confinement Area Ventilation System, DCHS bridge cranes, Lifting Fixtures
	Pre-closure Early Failure of Waste Package, No Filtration (NOTE 4)	Subsurface	Waste Package
	Criticality Due to Waste Package Internal Geometry Failure	DCHS Cell or Subsurface	Waste Package
	Criticality Due to Waste Package Flooding	DCHS Cell	Waste Package Design; no water pipes located in DCHS; waste package decontamination process does not utilize water
	Greater-than-6-Metric Ton Rock Fall on Waste Package, No Filtration	Subsurface – Emplacement Drift	Waste Package, Emplacement Drift Ground Control
	Greater-than-6-Metric Ton Rock Fall on Transporter, No Filtration	Subsurface – Main Drift	Waste Package, Waste Package Transporter, Locomotives
	Transporter Runaway, No Filtration	Subsurface – North Ramp	Waste Package Transporter

NOTE 1: This table, which is based on a similar table in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O 2000a), contains data that were updated during the development of the Repository Safety Strategy Revision 4.

NOTE 2: Unless "No Filtration" is shown, HEPA filters are assumed to be available in the event sequence.

NOTE 3: Exceptions are the naval canister and disposable commercial spent fuel canisters, which are not certified to withstand all credible handling impacts. The consequences of a naval canister breach are acceptable, i.e., the resulting offsite dose is below the regulatory limit. An analysis of a breach of a disposable commercial spent fuel canister has not yet been performed.

NOTE 4: Event consequences were analyzed deterministically (CRWMS M&O 2000a).

5. USE OF DESIGN MARGIN AND DEFENSE-IN-DEPTH IN PRECLOSURE REPOSITORY SYSTEM DESIGN AND OPERATIONS

5.1 INTRODUCTION

Because there will always be some uncertainty in estimates of repository preclosure safety, it is prudent not to rely solely on an integrated safety analysis to demonstrate reasonable assurance that performance standards will be satisfied. Confidence in the safety of the preclosure repository design is achieved by a combination of analyses of repository performance (including the ISA and analysis of beyond-design-basis events), application of the traditional concepts of margin and defense-in-depth, use of commercial nuclear industry and other nuclear fuel cycle facility precedent, and license specifications and surveillances.

Good engineering practice requires the design or operation of any SSC to include provisions that protect it from challenges or protect it from failure when subjected to challenges as well as contingencies that can be factored into the design to address failures of vital safety SSCs should they occur. This is almost always accomplished by ensuring sufficient margin is factored into the design of SSCs and/or through the application of defense-in-depth provisions.

5.2 DEFINITION OF DESIGN MARGIN

Use of margin is a standard engineering practice of including capability in excess of minimum design requirements in specifications for engineered SSCs and a standard operational practice of establishing conservative operating limits with respect to the minimum design requirements. This is required to provide for a reserve capability to account for any uncertainty in the conditions to which the SSCs will be subjected, to account for variability in the properties of component materials, and to account for the possibilities of overload or understrength due to variations in material dimensions, construction procedures, or simplifications of calculations.

The extent to which margin is incorporated in the design or operation of an SSC is not determined by rigid rules. Rather, engineers have traditionally established a level of margin based on several factors, including the importance to safety of an SSC, potential for severe consequences of failure, uncertainty about the magnitudes and characteristics of challenges, variability of material properties, potential for error in design or calculation, and cost. Questions of margin are commonly judged in terms of the degree of confidence that the response of the facility, a particular SSC, or an engineered barrier to postulated challenges is acceptable. The basis for the determination of what constitutes adequate margin derives primarily from engineering judgement based on experience. Just as there are no generally accepted rules for how much margin is enough in any particular situation, there also is no generally accepted process for describing how engineering judgement is exercised in any particular case to determine that a particular amount of margin is adequate.

Margin to conditions that might be detrimental to safety is established by choosing design criteria to be met for response of important to safety SSCs to various accidents and transients or postulated challenges to engineered barriers. These design criteria are calculated values of variables such as temperature or stress chosen to be sufficiently conservative to provide confidence that the uncertainties in component physical properties and other variability are bounded and thus provide margin to unacceptable conditions. Engineering analyses, evaluations,

and integrated safety analyses are performed to determine the response of SSCs and to demonstrate that the fundamental operating, regulatory, and safety criteria of the design are satisfied.

5.3 DEFINITION OF DEFENSE-IN-DEPTH

In the design and licensing of commercial nuclear facilities, defense-in-depth has evolved as a fundamental safety philosophy that employs successive measures in the form of diverse and redundant barriers or other protective measures to prevent or mitigate accidents and limit the consequences if a malfunction, accident, or the unexpected occurs. Defense-in-depth ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility (single failure criterion). A characteristic statement is, "Even if XXX improbably occurs, the consequences will be acceptable because YYY will prevent or mitigate...." The net effect of incorporating defense-in-depth into design, construction, maintenance and operation is that the facility or system in question tends to be more tolerant of failures and external challenges.

Defense-in-depth provides for the placement of multiple diverse and redundant physical and administrative barriers or other protective measures between the individual, public, or environment and the potentially harmful effects of radioactive materials. These barriers and protective measures compensate for uncertainty in system performance, human error, mechanical failures, uncertainty in the ISA calculations, and unanticipated characteristics of the engineered materials, SSCs. Table 5-1 illustrates the elements of defense-in-depth as they have evolved for power reactor design and licensing.

Table 5-1. Defense-in-Depth for Nuclear Power Plants

Physical Barriers	Administrative Barriers
• Non-urban siting • Fuel Oxide Matrix • Fuel Cladding • Primary Coolant Boundary—Piping, Vessels, etc. • Containment	• Conservatism, redundancy and diversity in design • Quality Assurance • Use of formal operating procedures and qualification of operators • Accident management (emergency plans and procedures) • Safety Culture

During the preclosure operational period of the repository, defense-in-depth is provided for in the traditional way by:

1) The selection of a remote site.
2) Use of physical or administrative barriers to reduce the frequency and/or consequences of potential design basis events;
3) Conservatism, redundancy, and diversity in repository facility design;
4) The application of QA to facility design, construction, operation and maintenance;
5) The imposition of radiation protection standards for workers and the public;
6) Formal operating procedures and qualification of operators; and

7) The establishment of emergency plans and procedures.

Table 5-2 illustrates the elements of defense-in-depth for repository preclosure safety.

Table 5-2. Defense-in-Depth for the Repository Preclosure Safety Case

Physical Barriers	Administrative Barriers
• Remote Desert Site	• Conservatism, Redundancy and Diversity in Design
• Fuel Oxide Matrix/Glass Matrix for HLW	• Quality Assurance
• Fuel Cladding/HLW and SNF Canister	• Use of Formal Operating Procedures and Qualification of Operators
• Shipping Cask/Disposal Container/Waste Package	• Technical Specifications and Surveillances
• Shipping Cask/Disposal Container/Waste Package Handling Systems and Lifting Fixtures	• Accident Management (emergency plans and procedures)
• Pool Storage Racks, Assembly Transfer Baskets, and Basket Staging Racks	• Safety Culture
• Pool Water Level Control Systems	
• Waste Handling Building Confinement	
• HEPA Filters	
• Waste Package Transporter/Locomotives	
• Underground Doors and Ventilation System	

5.4 RELATIONSHIP OF MARGIN AND DEFENSE-IN-DEPTH

Design margin and defense-in-depth are related concepts. Both are intended to provide a safety cushion against the unknown. The ISA and associated uncertainty analyses provide a risk-informed, performance-based framework for identifying where safety margin and/or defense-in-depth measures should be employed. Margin for individual physical barriers is provided through the specification of the physical and chemical properties of the barriers, e.g., size and/or material properties. In many instances, defense-in-depth is the primary methodology relied on for ensuring safety, usually through the incorporation of multiple barriers to failure. These barriers may be physical, such as metal canisters, impact limiters for shipping casks, HEPA filters in the heating and ventilation system, and redundant/diverse elements in SSCs important to safety. Barriers may also be procedural, such as use of a formal design control program, administrative, operating, and emergency procedures, and a quality assurance program.

Overall margin may also include defense-in-depth, especially when it is implemented by multiple barriers. Just as margin is added by making a barrier more robust (e.g., thicker), overall margin may also be increased by adding another barrier.

5.5 TRADITIONAL REACTOR ENGINEERING PRACTICE

Traditional reactor engineering practice relies heavily on both design margin and defense-in-depth to assure safety.

Design margin is typically factored into the design and fabrication process through a) the selection of conservative design and safety analysis input assumptions and acceptance criteria, and b) the application of appropriate engineering codes and standards in design and fabrication specifications in order to assure each SSC is functionally capable of performance in excess of minimum design requirements. Such standard engineering practice provides for a reserve capability that accounts for uncertainties, such as design analysis limitations, the range of conditions to which SSCs are subjected, or variability in the properties of component materials.

Defense-in-depth is incorporated through the use of multiple barriers to protect against the release of radioactive material to the environment. Reactor defense-in-depth physical barriers include features such as the solid ceramic fuel pellet matrix, fuel pin cladding, the reactor pressure vessel and coolant system piping, emergency core cooling systems, and reactor building containment systems. Each of these features represents a barrier that provides some degree of protection against radiological release in the event of reactor control system malfunctions. Each of these barriers is required to be designed and fabricated in a manner that provides margin to help assure performance of safety functions.

The extent to which margin and defense-in-depth is incorporated in traditional reactor engineering practices is primarily prescribed through regulatory requirements and industry codes and standards. Regulatory guidance documents were developed in parallel with industry codes and standards by expert panels to provide reactor designers and safety engineers with prescriptive design and safety analysis requirements. Expert panels relied on their knowledge and experience to derive reactor safety requirements, but they lacked significant risk analysis insights and thus generally relied heavily on margin and defense-in-depth concepts to address uncertainties that existed when developing regulations and standards. These deterministically derived prescriptive requirements included detailed specifications for required safety margins, defense-in-depth features, and safety analysis methodologies.

5.6 RISK-INFORMED DESIGN PROCESS

Risk-informed design practices have evolved that build upon traditional reactor engineering practice. Risk-informed design practices use risk insights in addition to traditional engineering practices to focus attention on the most important facility activities with respect to safety. Design criteria and management controls are established based upon the risk insights. Facility design is developed integral with a risk-based safety analysis in order to assure facility safety while minimizing design and operational complexity. Although a risk-informed design process makes use of risk-insights in the development of the facility design and design specifications, there continues to be attention to and consideration of traditional deterministic design guidance in the process. Continued attention to traditional deterministic design and regulatory guidance generally ensures that risk-informed designs will include comparable design margin as deterministically developed designs and will include appropriate defense-in-depth features. One significant distinction in the two design processes is the increased understanding of the risk reduction benefits of defense-in-depth features. The concept of blending risk insights with traditional deterministic design practices in the risk-informed design process is illustrated in Figure 5-1.

5.7 REPOSITORY SAFETY CASE

The ISA and the provision of margin and defense-in-depth are fundamental and essential elements of the repository safety case. These three elements are incorporated in the repository safety case as follows to:

- Achieve a reasonable balance between event prevention and consequence mitigation.
- Provide system redundancy, independence, and diversity commensurate with the expected frequency, consequences of challenges to the system, and uncertainties.

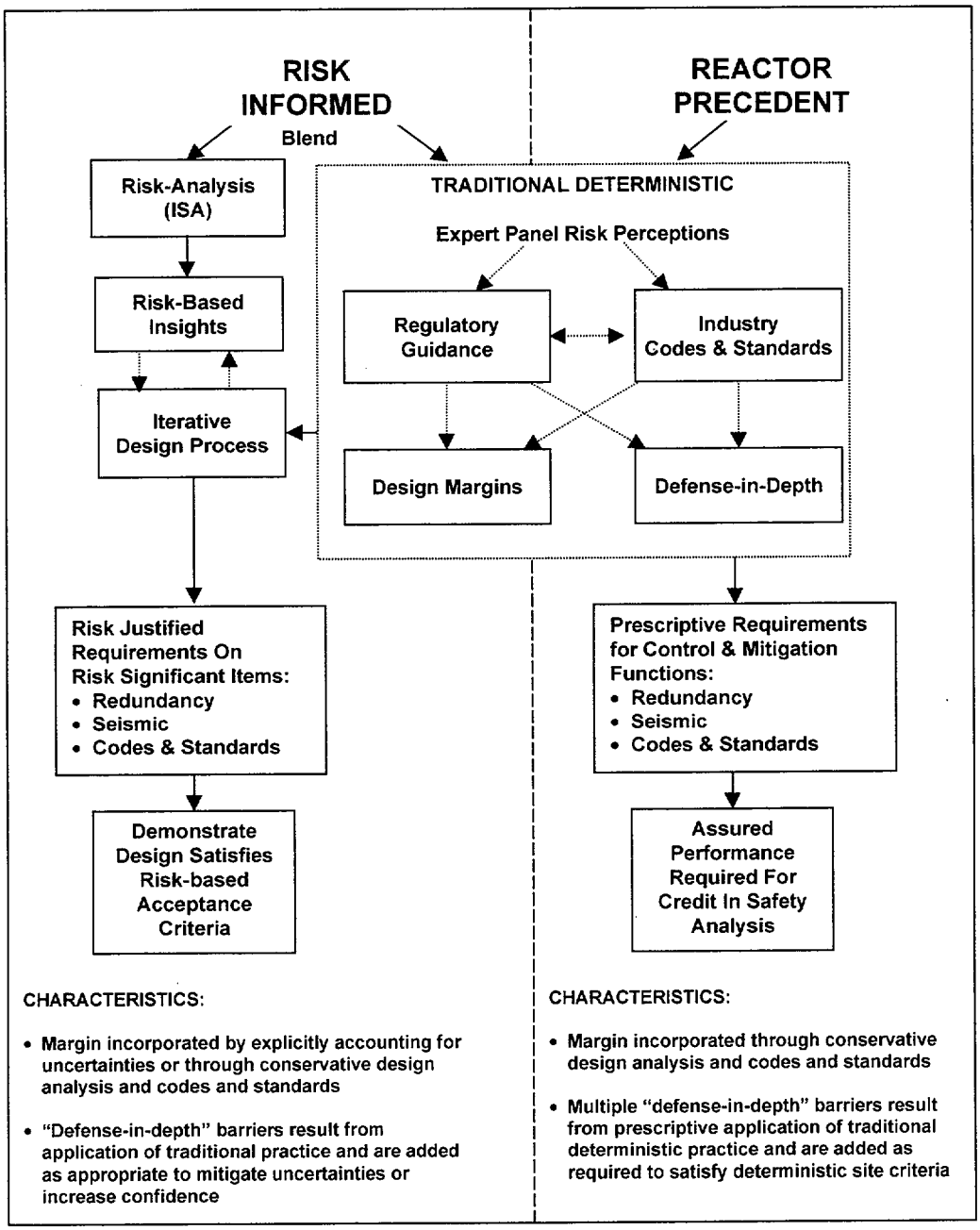

Figure 5-1. Application of Design Margin and Defense-in-Depth in a Risk-Informed Design Process

- Minimize the potential for common cause failures that can affect more than one defense-in-depth barrier.

- Avoid over-reliance on programmatic management controls to compensate for weaknesses in plant design.

Unlike a nuclear reactor, repository event sequences do not include failure of high-temperature and high-pressure fluid systems that can result in highly energetic releases of radioactive material. By contrast, even though large amounts of radioactive material and very high radiation fields are involved, event sequences for the MGR can be characterized as low-energy impacts that present virtually no mechanisms for releasing other than small, relatively insignificant amounts of radioactive material. Therefore, measures determined necessary for defense-in-depth will be commensurate with the risk significance of the kinds of event sequences that can occur during the MGR preclosure operations.

Defense-in-depth design features and management controls included in the repository preclosure safety case can be categorized into three types as follows:

- Inherent features for which no specific facility design feature or management control is required to implement a function, such as the physical form of the radioactive material allowed to enter the facility and the large distances to the controlled area boundary.

- Facility design features and management controls specifically identified by the ISA as required to satisfy 10 CFR 63 design basis event (DBE) frequency categories and/or 10 CFR 63.111 consequence limits (64 FR 8640), including consideration of uncertainties.

- Facility design features and management controls specifically required by 10 CFR 63, but which may not be explicitly identified by the Integrated Safety Analysis (ISA) as required to satisfy 10 CFR 63.111 DBE frequency and/or consequence limits (64 FR 8640), such as an emergency planning.

Specific defense-in-depth barriers to the release of radioactive materials to the environment include the fuel oxide matrix/glass matrix for HLW, fuel cladding/HLW canister, shipping cask/disposal canister/waste package, Waste Handling Building confinement systems, and a large distance to the controlled area boundary. The need for MGR facility defense-in-depth features or management controls that are not inherent to material physical form or site characteristics will be assessed within the framework of an ISA and the performance requirements established by 10 CFR 63.111. The necessity and sufficiency of safety measures will be determined based on the assessed margin(s) to the boundaries of 10 CFR 63 design basis event (DBE) frequency categories and/or 10 CFR 63.111 radiation exposure limits relative to the amount of uncertainty in the evaluations.

This approach to defense-in-depth evaluations for preclosure safety is in keeping with the NRC's philosophy expressed in Section VIII of the Supplementary Information included in the NRC's Notice of Proposed Rulemaking for 10 CFR 63 (64 FR 8640), wherein the NRC presents the basis for multiple barriers and defense-in-depth for the MGR. While maintaining that the application of the defense-in-depth concept is appropriate and reasonable, it states that

> Its implementation should be reexamined, in light of the advancement in methods to quantitatively assess the components of a geologic repository system and with consideration of requirements that are risk-informed and performance-based.

Most of the discussion focuses on defense-in-depth for postclosure considerations. For preclosure operations, the NRC states

> Its operation may be amenable to regulation comparable to other nuclear fuel cycle facilities licensed by NRC.

In this vein, proposed 10 CFR 63.112 (e), outlines information to be considered in the ISA to identify "measures taken to ensure the availability of identified safety systems." Such measures, including means to provide redundant systems, emergency power, means to test and inspect, provide defense-in-depth as appropriate for the degree of uncertainty that is established for the ISA models and parameters.

The ISA results, with consideration of uncertainty and conservatism already credited in the analysis for a given accident scenario or event sequence, will provide the basis for identifying the degree of defense-in-depth present in the design, or the need for additional defense-in-depth measures, that may include one or more of the following:

- Redundant and/or diverse physical barriers to contain releases of radioactive material, or to dilute a release (e.g., multiple levels of filtration).
- Redundant and/or diverse subsystems including prevention and/or mitigation measures; e.g., single-failure-proof, nuclear-grade cranes and manipulators.
- Redundant fans/motors/power supplies and/or HEPA banks in the WHB ventilation system.
- Redundant and diverse control and brake systems on waste-package transporter.

Consistent with traditional engineering practice, each of the engineered barriers identified as necessary in the ISA will be designed and fabricated in a manner that provides margin to assure performance of safety functions. Uncertainties in performance reliability may also be explicitly evaluated within the context of the preclosure safety analysis and defense-in-depth evaluations.

Several sources of uncertainty will be addressed in the preclosure safety analysis and defense-in-depth evaluations, for example:

- uncertainties in event sequence identification and modeling;
- uncertainties in the parameters used to estimate the frequencies and probabilities of initiating events and system unavailability used in event sequence frequency analyses;
- uncertainties in estimates of the amounts of radioactivity that are released into various pathways (e.g., the airborne and respirable fractions) in various accident scenarios; and

- uncertainties in the parameters used for evaluating the concentrations of radionuclides that reach hypothetical members of the public at or beyond the controlled area boundary.

To the extent possible, the ISA will include quantitative analyses of the uncertainties, and decisions on defense-in-depth will be made accordingly.

6. STRATEGY FOR PREVENTION OR MITIGATION OF PRECLOSURE OFFSITE RADIATION EXPOSURE

6.1 IDENTIFICATION OF IMPORTANT TO RADIOLOGICAL SAFETY FEATURES AND CONTROLS

To ensure radiation doses associated with Category 1 and Category 2 design basis events do not exceed the limits in proposed 10 CFR Parts 63.111(a)(2) and 63.111(b)(2), the repository design will incorporate a combination of prevention and mitigation features and controls (64 FR 8640). Prevention is the use of design features to reduce the postulated frequency of events that result in radiological release from the GROA to less than one chance in 10,000 of occurring before permanent closure. Mitigation is the utilization of design features/barriers to ensure the consequences of a postulated radiological release event sequence are within the respective regulatory limits for doses to workers and the public. Mitigation includes those features intended to reduce releases from routine operations that are included in the Category 1 events annual dose summation. The ISA is used to identify preventive and mitigative features and operational controls needed to demonstrate compliance with radiation dose limits.

This strategy uses prevention features wherever practical. By eliminating or minimizing the potential for radiological release events, this strategy provides operational benefits. From an operations perspective, surveillance and maintenance of active safety features for mitigating the consequences of events have been demonstrated to add significantly to nuclear facility operational complexity, and recovery from events has proved to be significantly more challenging than anticipated. This strategy is implemented by performing the ISA as an integral part of the design process in a manner consistent with a performance-based, risk-informed philosophy. A risk-informed approach uses risk insights, engineering analysis and judgement, and equipment performance history to focus attention on the repository preclosure operations functions that have the most safety significance and to establish design criteria and management controls based upon these risk insights. This integral design approach assures that design features and operational controls important to radiological safety are selected in a manner that assures safety, while minimizing design and operational complexity through the use of proven technology.

Loading and unloading of spent nuclear fuel transportation casks are done on a routine basis in the United States. For example, from 1979 to 1997, some 1,300 shipments of SNF were made by the commercial nuclear industry (NRC 1998, Table 3.2), and from 1957 to 1998, some 700 shipments of spent nuclear fuel were made by the U.S. Navy (Beckett, T.H. 1998, Attachment 2, Table 1). This represents some 4,000 loadings or unloadings of spent nuclear fuel transportation casks. Heavy loads are routinely moved by bridge cranes at nuclear facilities, as they would be at the repository. Transportation of heavy equipment by rail is also commonly done underground in the mining industry. Across the United States, commercial nuclear power facilities currently operate 110 spent nuclear fuel pools; handling of SNF is a routine operation at these facilities. For example, from 1968 to 1994, about 105,000 spent nuclear fuel assemblies were discharged from commercial nuclear power facilities in the United States (DOE 1996, Table 5). This means that, at U. S. commercial facilities alone, SNF assemblies have been safely handled at least 105,000 times—or an average of about 3,900 times per year. Decontamination activities and processing of low-level radioactive waste are also routinely performed at

commercial nuclear power and DOE facilities. The lessons learned from these experiences will be incorporated into the design and concept of operations for the potential repository.

6.2 DESIGN BASES FOR FACILITIES AND LIMITS ON OPERATIONS

The preclosure safety strategy is to design, construct, and operate SSCs, that are important to safety, so that they survive credible external events and natural phenomena in a manner that the Category 1 and Category 2 design basis event dose limits are not exceeded. For example, the Waste Handling Building will be designed to withstand the ground motions associated with earthquakes. Likewise the spent fuel assembly racks in the assembly transfer system will be designed to withstand earthquakes. For the assembly transfer and canister transfer systems, overhead cranes and assembly transfer machines will be designed so they will not become dislodged from their rails during earthquakes.

For accidents involving internal events, event sequences involving SNF assembly and canister drops are important contributors to potential offsite dose. The assembly transfer system will be designed, constructed, and operated so the probability of the dry assembly transfer machine dropping an assembly is very low and the probability of a drop of a spent fuel assembly onto another spent fuel assembly is limited. Likewise, the canister transfer system will be designed, constructed, and operated so the probability of a crane dropping a canister is very low. In addition, lift heights will be limited, as is standard practice in nuclear facility design, and canisters will be tested to ensure that these drops do not result in a breach of the canister containment boundary unless it can be demonstrated that releases from a breached container result in site boundary doses below the regulatory limits without taking credit for the HEPA filters in the ventilation system. The consequences of a breach of the disposable canisters containing Navy spent fuel are acceptable, i.e., the resulting offsite dose is below the regulatory limit. The consequences of a breach of a DOE or commercial spent fuel canister have not been analyzed using fully qualified data. The waste packages will be designed to survive the drop event.

The analyses in *Design Basis Event Frequency and Dose Calculation for Site Recommendation* (CRWMS M&O 2000b, Section 5.2.5) also showed that the availability of the Waste Handling Building ventilation system plays a large role in mitigating the consequences of accidents. Therefore, the ventilation system will be designed, constructed, and operated to be highly reliable. For example, ventilation system components required to perform a safety function will have the capability to withstand design basis earthquakes, tornadoes, fires, and loss of offsite power.

6.3 SAFETY STRATEGY FOR REPOSITORY PRECLOSURE OPERATIONS FUNCTIONS

This strategy is based on the repository preclosure operations functions: (1) receipt of waste; (2) transfer of waste into the disposal container; (3) sealing the disposal container; (4) transfer of the waste package to the emplacement drift; (5) emplacement; and (6) monitoring until repository closure. The strategy for each of the basic functions is either prevention augmented by mitigation or mitigation augmented by prevention. The safety strategy for the first five of the repository preclosure operations functions and the list of safety features for each was reported in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site*

Recommendation (CRWMS M&O 2000a) and are summarized in Table 6-1 and discussed below. A detailed safety assessment for the monitoring period has yet to be performed, however, it is anticipated that the relevant DBEs, safety strategy, safety features, and calculated offsite exposures will be bounded by those determined from evaluation of the first five functions and reported in (CRWMS M&O 2000a).

Table 6-1. Repository Preclosure Safety Strategy

Receipt of Waste				
Basic Operations	**Canistered Fuel and HLW**		**Uncanistered fuel**	
	Safety Strategy	**Safety Systems**	**Safety Strategy**	**Safety Systems**
Survey Remove impact limiters Remove personnel barriers Remove hold downs Upright cask Transfer cask to cart	Prevent events that could exceed shipping cask design basis (preclude breach)	Shipping Cask	Prevent events that could exceed shipping cask design basis (preclude breach)	Shipping Cask
Transfer of Waste into the Disposal Container				
Basic Operations	**Canistered Fuel and HLW**		**Uncanistered fuel**	
	Safety Strategy	**Safety Systems**	**Safety Strategy**	**Safety Systems**
Vent/Sample cask Unbolt cask cover Remove cover Remove waste/SNF from cask Install cover Bolt cask cover Store canistered waste Store SNF assemblies Decontaminate cask Remove DC cover Load DC Install DC cover Decontaminate DC	Prevent events that could exceed canister design basis (preclude breach) (Note 1)	Canister	Minimize the number of events that could result in uncanistered fuel drops; minimize releases of radioactive material from drop events	WHB Confinement
Sealing the Disposal Container				
Basic Operations	**Canistered Fuel and HLW**		**Uncanistered fuel**	
	Safety Strategy	**Safety Systems**	**Safety Strategy**	**Safety Systems**
Weld DC Inspect DC welds Stress relieve DC welds	Prevent events that could exceed canister design basis (preclude breach)	Canister	Minimize the number of events that could result in disposal container drops; minimize releases of radioactive material from drop events	WHB Confinement

Table 6-1. Repository Preclosure Safety Strategy (Cont'd.)

Transfer of the Waste Package to the Emplacement Drift				
Basic Operations	**Canistered Fuel and HLW**		**Uncanistered fuel**	
	Safety Strategy	**Safety Systems**	**Safety Strategy**	**Safety Systems**
Load WP onto WP Transporter Move WP and pallet to tunnel entrance Descent to drift entrance Park at drift entrance	Prevent events that could exceed WP design basis (preclude breach)	WP, WP Transporter	Prevent events that could exceed WP design basis (preclude breach)	WP, WP Transporter
Emplacement				
Basic Operations	**Canistered Fuel and HLW**		**Uncanistered fuel**	
	Safety Strategy	**Safety Systems**	**Safety Strategy**	**Safety Systems**
Move WP and pallet from drift entrance to permanent drift position	Prevent events that could exceed WP design basis (preclude breach)	WP	Prevent events that could exceed WP design basis (preclude breach)	WP

NOTE 1: For Navy fuel and HLW, credit can be taken for waste form integrity. For commercial and DOE SNF, canisters need not be credited if releases from a breached canister result in site boundary doses below the regulatory limits without taking credit for the HEPA filters in the ventilation system.

Receipt of Waste–This function covers operations from the time a transportation cask containing spent fuel or HLW arrives at the repository up to the time the transportation cask is opened. Containment is provided by the transportation cask. Prevention of radiological releases from the GROA is provided by the transportation cask containment features, design of the repository surface facility cask handling features, and controls that preclude events that could result in exceeding the cask design basis for containment integrity, such as those provided to limit cask lift height.

Transfer of Waste into the Disposal Container–This function covers opening the transportation cask, handling and transfer of waste from the open transportation cask to the disposal container and transfer of the disposal container to the welding area. Prevention is provided by the surface facility design features and controls that reduce the likelihood of unsealed transportation cask drop events. For commercial SNF, spent fuel pool water and Waste Handling Building and transfer cell confinement systems provide mitigation functions. For small disposable DOE SNF and HLW canisters, radiological release prevention is provided by canister containment features, augmented by canister transfer system features and controls that prevent events that could result in exceeding canister design basis for containment integrity, such as those provided to limit canister lift height. For large, disposable canisters (e.g., naval spent fuel canisters and commercial SNF canisters for which it can be demonstrated that releases from a breach result in site boundary doses below the regulatory limits without taking credit for the HEPA filters in the ventilation system), the Waste Handling Building (Canister Transfer System) provides a mitigation function.

Sealing the Disposal Container–This function covers the handling of the loaded disposal container from the time it first arrives in the welding area to the time the loaded, closed, welded, inspected, and certified disposal container (now called a waste package) along with its emplacement pallet, is placed into the transporter. For uncanistered SNF and large disposable canisters, prevention is provided by the surface facility features and controls that reduce the likelihood of drop events, and mitigation is provided by transfer cell confinement system features. For small disposable canisters, prevention is provided by the sealed canister containment features, augmented by facility features and controls that prevent events that could result in exceeding the canister design basis for containment integrity, such as features and controls provided to limit canister lift height. Whatever its contents, the waste package will provide containment and serve as a preventive feature, augmented by surface facility cask handling features and controls that prevent events that could result in exceeding the waste package design basis for containment integrity.

Transfer of the Waste Package to the Emplacement Drift–This function covers the handling of a waste package and emplacement pallet from the time they are loaded into the transporter through the time the transporter parks at the entrance to the designated emplacement drift. Before the transporter's descent via the ramp into the repository's subsurface, the waste package will provide containment and serve as a preventive feature, augmented by the transporter and rail system that will prevent events that could result in exceeding the waste package design basis for containment integrity. During descent on the ramp, the waste package will provide containment and serve as a preventive feature, augmented by the transporter's safety features, such as the brakes and the rail system, that prevent events that could result in exceeding the waste package design basis for containment integrity. Other safety features, such as the use of dual locomotives to move the transporter, also perform preventive functions during descent. During parking at the emplacement drift, the waste package will provide containment and serve a preventive function, augmented by the transporter and the rail and ground support systems to ensure that no credible events can occur that are beyond the waste package design basis for containment integrity.

Emplacement–This function covers handling and moving the waste package and emplacement pallet from the time when the transporter is safely parked at the entrance to the emplacement drift through the final emplacement of the waste package and emplacement pallet in their permanent position. The waste package will provide containment and serve as a preventive feature, augmented by the transporter, rail system, pallet, and ground support to ensure that no credible events can occur that are beyond the waste package design basis for containment integrity.

Monitoring Until Repository Closure–This function consists of monitoring the emplaced waste to detect abnormalities or early failure, surveillance and performance of any necessary maintenance on the emplacement drifts and ventilation systems, and ongoing Performance Confirmation testing and analyses until a decision is make to close the repository. This function, like all the other functions, would be performed by the Department of Energy under the regulatory oversight of NRC.

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7. DESIGN BASIS EVENT CONSEQUENCE ANALYSIS

The report *Design Basis Event Frequency and Dose Calculation for Site Recommendation* (CRWMS M&O 2000b) evaluated the consequences of Category 1 and Category 2 DBEs for the Site Recommendation Consideration Report (SRCR) reference repository design. The results are summarized in Table 7-1.

For Category 1 DBEs and normal operations releases, offsite radiation doses were based on the following:

- Radiation doses for the sum of anticipated releases and Category 1 DBEs weighted by the annual frequency of each event
- Inhalation, ingestion, submersion in air, and external exposure pathways
- Mitigation by HEPA filters of particulate releases from the surface facilities
- Distances of 11 km (7 mi) from the Waste Handling Building to the offsite boundary for surface facility releases and 8 km (5 mi) from the repository to the offsite boundary for subsurface releases (CRWMS M&O 2000b, Section 3.3)
- Annual average ground level atmospheric dispersion factors (CRWMS M&O 2000b, Section 5.2.2).

The sum of all radiation doses[7] to an average member of the public conservatively estimated for anticipated radioactive effluent releases and Category 1 DBEs was 0.06 mrem/yr (CRWMS M&O 2000a, Section 5.3.6.1). This is well below the 25 mrem/yr limit specified in proposed 10 CFR Part 63.111(a)(2) (64 FR 8640), the 15 mrem/yr limit specified in proposed 40 CFR Part 197.4(a) (64 FR 46976), the 100 mrem/yr limit specified in 10 CFR Part 20.1301(a)(1), and the 10 mrem/yr constraint specified in 10 CFR Part 20.1101(d) for implementation of the ALARA requirement for air emissions of radioactive material to the environment. In addition, the external radiation dose to the public is expected to be orders of magnitude lower than the 2 mrem/hr limit specified in 10 CFR Part 20.1301(a)(2).

For workers, the sum of the radiation doses from anticipated radioactive effluent releases and Category 1 DBEs was estimated to be 0.01 rem/yr TEDE per exposed worker (CRWMS M&O 2000a, Section 5.3.6.2), well below the 5 rem/yr limit specified in 10 CFR Part 20.1201(a)(1). The largest radiation dose to any organ or tissue other than the lens of the eye was estimated to be 0.1 rem/yr (CRWMS M&O 2000a, Section 5.3.6.2), well below the 50 rem/yr limit specified in 10 CFR Part 20.1201(a)(1). The radiation dose to the skin was estimated to be 0.1 rem/yr (CRWMS M&O 2000a, Section 5.3.6.2), well below the 50 rem/yr limit specified in 10 CFR Part 20.1201(a)(2). These radiation doses were based on the inhalation and external exposure pathways at a distance of 100 m (328 ft). This distance is typical of nuclear facility dose calculations for noninvolved or collocated workers (e.g., DOE 1999, p. 4-13). Radiation dose coefficients are not available in *External Exposure to Radionuclides in Air, Water, and Soil*

[7] Some regulatory dose limits are based on the total effective dose equivalent (TEDE) in a year; however, the limit has been expressed as dose per year (e.g. mrem/yr) for simplicity.

(Eckerman, 1993) for the lens of the eye. However, the *NRC Standard Review Plan for Spent Fuel Dry storage Facilities*, NUREG-1567 (NRC 2000, Section 9.5.2.2) states that compliance with the radiation dose limit to the lens of the eye is achieved if the sum of the skin dose and the TEDE does not exceed 15 rem. For anticipated radioactive effluent releases and Category 1 DBEs the sum of these two doses (0.01 rem + 0.10 rem = 0.11 rem) is well below the 15 rem limit specified in 10 CFR Part 20.1201(a).

The radiation dose for workers exposed to radioactivity during normal waste handling operations has not been estimated but is expected to be a small fraction of the 5 rem/yr limit since these exposures are largely controllable (e.g., minimized by shielding design features and remote operations) and will be maintained ALARA.

For Category 2 DBEs, offsite radiation doses were based on the following :

- Conservative values of exposures on a per event basis
- Inhalation and external exposure pathways
- Release fractions that include credit for the respirable fraction of particulates
- Mitigation by HEPA filters of particulate releases from the surface facilities
- Distances of 11 km (7 mi) from the Waste Handling Building to the offsite boundary for surface facility releases and 8 km (5 mi) from the repository to the offsite boundary for subsurface releases (CRWMS M&O 2000b, Section 3.3)
- 99.5 percentile ground level atmospheric dispersion factors (CRWMS M&O 2000b, Section 5.2.2).

The maximum radiation dose to a member of the public for a Category 2 DBE was conservatively calculated to be 0.02 rem (TEDE) (CRWMS M&O 2000a, Section 5.3.6.1). This is well below the 5 rem per event sequence limit in proposed 10 CFR Part 63.111(b)(2) (64 FR 8640). The largest radiation dose to any organ or tissue other than the lens of the eye was estimated to be 0.1 rem (CRWMS M&O 2000a, Section 5.3.6.1), well below the 50 rem limit specified in proposed 10 CFR Part 63.111(b)(2). The radiation dose to the skin was estimated to be 0.04 rem (CRWMS M&O 2000a, Section 5.3.6.1), well below the 50 rem limit specified in proposed 10 CFR Part 63.111(b)(2). Radiation dose coefficients are not available in *External Exposure to Radionuclides in Air, Water, and Soil* (Eckerman, 1993) for the lens of the eye. However, the *NRC Standard Review Plan for Spent Fuel Dry storage Facilities*, NUREG-1567 (NRC 2000, Section 9.5.2.2) states that compliance with the radiation dose limit to the lens of the eye is achieved if the sum of the skin dose and the TEDE does not exceed 15 rem. For the maximum radiological consequence Category 2 DBE the sum of these two doses (0.02 rem + 0.04 rem = 0.06 rem) is well below the 15 rem limit specified in proposed 10 CFR Part 63.111(b)(2).

Table 7-1. Preclosure Dose Calculation Results for SRCR Reference Design

DBEs	Dose	Regulatory Limit (Note 1) (Note 2)		Maximum Estimated Dose With HEPA Filters	Safety Factor
Category 1 Events	Offsite (Public) Dose	10 CFR Part 63.111(a)(2)	25 mrem per year	0.06 mrem TEDE per year	400
	Offsite (Public) Dose	40 CFR Part 197.4(a)	15 mrem per year	0.06 mrem TEDE per year	250
	Offsite (Public) Dose	10 CFR Part 20.1301(a)(1)	100 mrem per year	0.06 mrem TEDE per year	1500
	Offsite (Public) Dose	10 CFR Part 20.1101(d)	10 mrem per year	0.06 mrem TEDE per year	150
	Worker Dose	10 CFR Part 20.1201(a)(1)	5 rem per year	0.01 rem TEDE per year	500
	Organ/Tissue Dose	10 CFR Part 20.1201(a)(1)	50 rem per year	0.1 rem per year	500
	Skin Dose	10 CFR Part 20.1201(a)(2)	50 rem per year	0.1 rem per year	500
Category 2 Events	Offsite Dose	10 CFR Part 63.111(b)(2)	5 rem per event	0.02 rem TEDE per event	250
	Organ/Tissue Dose	10 CFR Part 63.111(b)(2)	50 rem per event	0.1 rem per event	500
	Skin Dose	10 CFR Part 63.111(b)(2)	50 rem per event	0.04 rem per event	1250

Note 1: Some regulatory dose limits are based on the total effective dose equivalent (TEDE) in a year; however, the limit has been expressed as dose per year (e.g. mrem/yr) for simplicity.

Note 2: The regulatory dose limits are taken from the NRC proposed 10 CFR Part 63 and the EPA proposed 40 CFR Part 197.

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8. QUALITY CLASSIFICATION ANALYSIS

The ISA provides input to the quality assurance classification process. Repository features given credit for event prevention or mitigation in the ISA fall within the proposed 10 CFR Part 63.2 definition of important to safety (64 FR 8640). The ISA is the tool used to determine an item's functional role as part of the repository preclosure safety case. SSCs important to safety are classified in a graded fashion to assure quality assurance controls are implemented over the facility life cycle commensurate with an item's importance to safety. Features important to safety are classified based on a hierarchy of safety significance.

The classification process considers the configuration and function of SSCs and their effect on repository radiological safety. The classification analyses are based on the system design and functions as established by the System Description Documents. These analyses utilize the results of the DBE analysis to evaluate the repository preclosure operations facility SSCs against the classification criteria of procedure QAP-2-3 to determine the quality assurance classification of the particular item (See Figure 8-1). The classification categories are specified to meet the requirements of Section 2 of the *Quality Assurance Requirements and Description* (QARD) (DOE 2000). SSCs classified according to quality level (QL) are subject to the requirements of the QARD.

8.1 QUALITY LEVEL 1 (QL-1)

QL-1 SSCs are those determined to be important to safety, whose failure could *directly* result in a condition adversely affecting public safety. These items have a high safety or waste isolation significance.

For preclosure safety considerations, QL-1 SSCs include those items required to:

- Maintain waste package containment or criticality control for SNF and high-level radioactive waste or
- Prevent or mitigate a Category 1 or 2 DBE that could result in exceeding specified offsite dose limits.[8]

Based on the criteria above the following are examples of SSCs that are expected to be classified as QL-1:

- Waste packages which provide containment and criticality control for commercial SNF assemblies;
- Disposable canisters which provide containment for DOE high level waste and DOE SNF
- The Waste Handling Building structure, which provides confinement for radioactive materials, radiation shielding, and protection of equipment from internal and external hazards; and

[8] In the "MGR Quality Level 1 Checklist" contained in *Classification of Permanent Items* (QAP-2-3, Rev. 10, Attachment II), the QL-1 dose limit for Category 1 design basis events is 100 mrem/yr.

- Locomotives, which are equipped with features to prevent uncontrolled descent of the waste package transporter and the possible impact of a waste package with the subsurface facility structure or other facility equipment resulting in radiological release.

8.2 QUALITY LEVEL 2 (QL-2)

QL-2 SSCs are those that have been determined to be important to safety and whose failure or malfunction could *indirectly* result in a condition adversely affecting public safety, or whose *direct* failure would result in consequences in excess of normal operational limits. These items have a lower safety or waste isolation significance.

QL-2 SSCs include those items required to:

- Provide control and management of site-generated radioactive waste;
- Provide fire protection/suppression to protect the function important to safety of a SSC specified as QL-1;
- Maintain SSC integrity so as not to prevent a QL-1 SSC from performing its intended function in the event of a DBE;
- Prevent or mitigate a Category 1 DBE that could result in exceeding specified offsite dose limits[9]; or
- In conjunction with an additional item or administrative control, prevent or mitigate a Category 1 or Category 2 DBE that could result in exceeding specified offsite dose limits.

Based on the criteria above the following are examples of SSCs that are expected to be classified as QL-2:

- Bridge cranes, which are designed to maintain structural integrity in the event of a DBE, e.g., seismic event so as to prevent interactions with QL-1 SSCs;
- Waste Handling Building (WHB) Fire Suppression Systems, which protect QL-1 SSCs from the effects of fire; and
- WHB ventilation systems, which mitigate the consequences of DBEs.

8.3 QUALITY LEVEL 3 (QL-3)

QL-3 SSCs are those that have been determined to be important to safety, but whose failure or malfunction would not *significantly* impact public or worker safety, including those defense-in-depth design features intended to keep radiation doses ALARA. These items have a minor impact on public and worker safety and waste isolation.

[9] In the "MGR Quality Level 2 Checklist" contained in *Classification of Permanent Items* (QAP-2-3, Rev. 10, Attachment II), the QL-2 dose limit for Category 1 design basis events is 25 mrem/yr.

QL-3 SSCs include those items required to:

- Provide warning of significant increases in radiation levels or concentrations of radioactive materials;
- Monitor variables to verify that operating conditions are within technical specification limits;
- Support repository emergency response actions;
- Assess radionuclide release or dispersion following a DBE;
- Maintain levels of radioactive material in facility effluents ALARA; or
- Limit worker doses from normal operations and Category 1 DBEs.

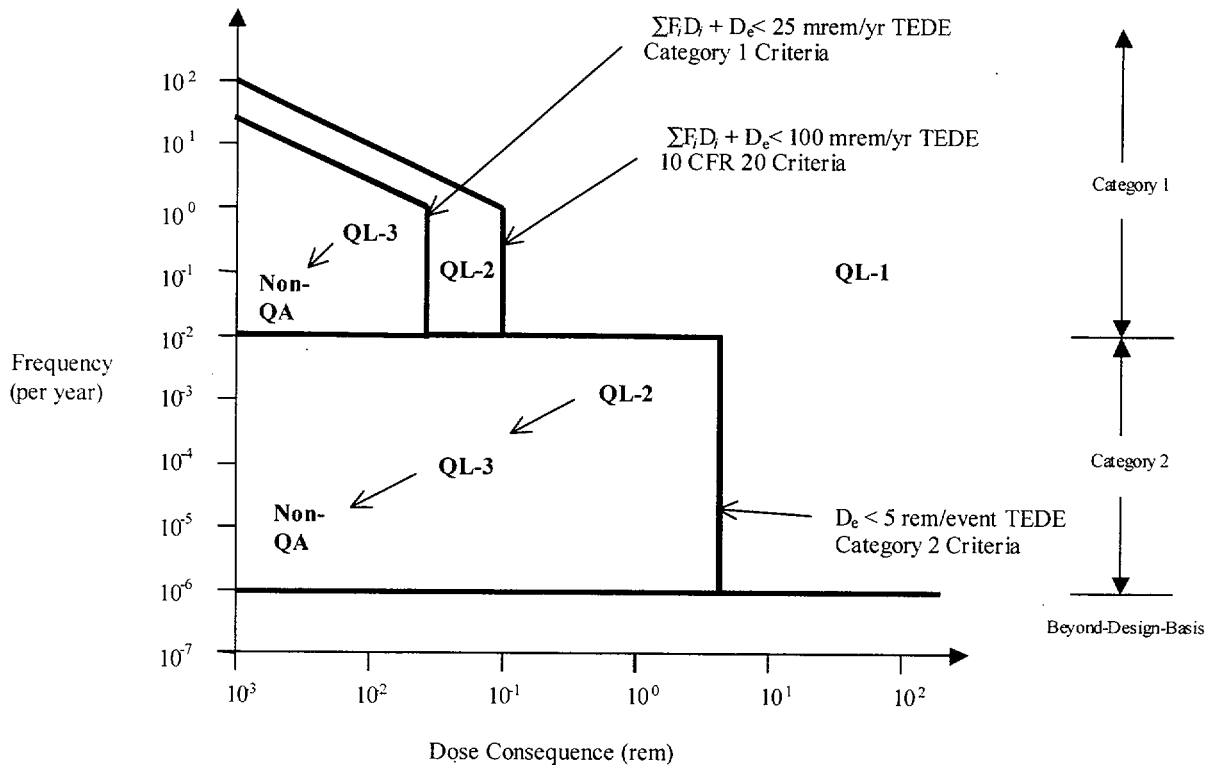
Based on the criteria above the following are examples of SSCs that are expected to be classified as QL-3:

- the area radiation monitoring system which provides an alarm to personnel to warn of significant increases to radiation levels in the various operating areas;
- the WHB exhaust stack radioactivity monitor which provides an alarm to warn of increases in the concentrations of airborne radioactive materials in the effluent discharged to the environment; and
- pool water leak detection and pool water level control systems, which monitor the pool level to verify it remains within license specification limits.

8.4 CONVENTIONAL QUALITY (CQ)

CQ SSCs are those not meeting any of the criteria for QL-1, QL-2, or QL-3. CQ items are not subject to the requirements of *the Quality Assurance Requirements and Description* (DOE 2000). Some CQ SSCs are important to the DOE program due to efficiency, economy, and production rate criteria, even though they are not associated with the QL type of considerations.

The classification of SSCs is illustrated by Figure 8-1. A simple take-away analysis is done where the effect of not taking credit for each safety-related SSC is determined one at a time. If the 100 mrem/year Category 1 or 5 rem/event Category 2 dose limit is exceeded, the SSC is QL-1. If the QL-1 dose limits are not exceeded, but the Category 1 dose limit of 25 mrem/year is exceeded the SSC is QL-2. If no dose limit is exceeded the SSC can be QL-2 or QL-3 depending on its function. For example, if the HEPA filters are not credited for reducing the concentration of radioactive particulates in the exhaust of the WHB ventilation system, the offsite dose for certain Category 2 events increases from 0.02 rem to 2.0 rem, but remains below the limit. The HEPA filters are not required to meet the offsite dose limit. Therefore, with respect to mitigation of Category 2 events the HEPA filters could be classified as a QL-2 system, i.e., a system that provides defense-in-depth.



Where: F_i = annual frequency (per year) of the i th event;
D_i = annual dose from i th event (rem);
D_e = dose from bounding event (rem or rem per year); and
$\Sigma F_i D_i + D_e$ = the annual average radiation dose for the sum of anticipated releases and the bounding Category 1 design basis event

Figure 8-1. Quality Level Classification Criteria

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9.2 CODES, STANDARDS, AND REGULATIONS

10 CFR 20. Energy: Standards for Protection Against Radiation. Readily Available.

64 FR 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Readily Available.

64 FR 46976. Environmental Radiation Protection Standards for Yucca Mountain, Nevada; Readily Available.

APPENDIX A

PRECLOSURE SAFETY STRATEGY GLOSSARY

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APPENDIX A

PRECLOSURE SAFETY STRATEGY GLOSSARY

Backfill	The material (e.g., tuff gravel, sand) placed in the excavated areas of the underground facility.
Barrier	Any material or structure that prevents or substantially delays movement of water and/or radionuclides (Proposed 10 CFR Part 63.2). In this volume a barrier is also any device or measure that decreases the likelihood of occurrence or adverse effects of a threat to safety or quality.
Breach	An opening in the transportation cask, SNF canister, disposal canister, waste package or drip shield caused by corrosion or mechanical stress.
Cladding	The metallic outer sheath of a fuel element.
Commercial Spent Nuclear Fuel	Commercial nuclear fuel elements that have been removed from a nuclear reactor following irradiation.
Containment	The confinement of radioactive waste within a designated boundary.
Credible Event	An event or event sequence having a probability of occurrence of at least 1 in 10,000 prior to the final closure of the repository.
Criticality	The condition in which nuclear fuel sustains a chain reaction. It occurs when the number of neutrons present in one generation cycle equals the number generated in the previous cycle. The state is considered critical when a self-sustaining nuclear chain reaction is ongoing.
Defense-in-Depth	The term used to describe the property of a system of multiple barriers to mitigate unanticipated conditions, processes, and events such that the system does not rely unduly on any single barrier.

Design Bases	That information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or requirements. These requirements are derived from analysis (based on calculation or experiments) of the effects of a postulated event under which a structure, system, or component must meet its functional goals. The values for controlling parameters for external events include: 1) Estimates of severe natural events to be used for deriving design bases that will be based on consideration of historical data on the associated parameters, physical data, or analysis of upper limits of the physical processes involved; and 2) Estimates of severe external human-induced events, to be used for deriving design bases that will be based on analysis of human activity in the region, taking into account the site characteristics and the risks associated with the event.
Design Bases Events	1) Those natural and human-induced events that are expected to occur one or more times before permanent closure of the geologic repository operations area (referred to as Category 1 events); and 2) Other natural and man-induced events that have at least one chance in 10,000 of occurring before permanent closure of the geologic repository (referred to as Category 2 events).
Disposable Canister	A containment vessel structure licensed for dry storage, transportation, and subsequent disposal in the repository. This type of canister can be placed directly into a disposal container and eliminates the need to open the canister and repackage individual fuel assemblies once it has been loaded for storage. It is expected that defense high-level radioactive waste forms and most DOE-owned spent nuclear fuel will be received in disposable canisters. Formally referred to as a dual-purpose canister or multi-purpose canister.
Disposal Container	The vessel consisting of the barrier materials and internal components in which the canistered or uncanistered waste form is placed. The disposal container is designed to contain spent nuclear fuel and high-level radioactive waste, but exists only until the outer lid weld is complete and accepted. Once it has been loaded with waste, the outer lid welded in place and the welds pass the certification inspection the disposal container is thereafter referred to as the waste package, which is the name of the vessel emplaced underground.
DOE Spent Nuclear Fuel	Radioactive waste created by defense activities.

Dose	Radiation dose is the quantity of radiation energy absorbed in a material. There are several terms used in radiation protection to precisely describe the various aspects associated with the concept of dose and how radiation energy deposited in tissue affects humans, e.g. rem, millirem, Sievert.
Drift	A horizontal passage in the underground facility.
Geologic Repository	A system that is intended to be used for or may be used for, the disposal of radioactive wastes in excavated geologic media. A geologic repository includes: The engineered barrier system, and the portion of the geologic setting that provides isolation of the radioactive waste.
Geologic Repository Operations Area	A high-level radioactive waste facility that is part of a geologic repository, including both surface and subsurface areas, where waste handling activities are conducted.
Important to Safety	With reference to structures, systems, and components, those engineered features of the geologic repository operations area, whose function is: 1) To provide reasonable assurance that high-level waste can be received, handled, packaged, stored, emplaced, and retrieved without exceeding the requirements of proposed 10 CFR Part 63.111(b)(1) for Category 1 design basis events; or 2) To prevent or mitigate Category 2 design basis events that could result in doses equal to or greater than the values specified in proposed 10 CFR Part 63.111(b)(2) to any individual located on or beyond any point on the boundary of the site.
Integrated Safety Analysis (ISA)	An analysis to identify hazards and their potential for initiating event sequences, the potential event sequences and their consequences, and the site, structures, systems, components, equipment, and activities of personnel, that are relied on for safety. As used here, integrated means joint consideration of safety measures that otherwise might conflict, including, but not limited to, integration of fire protection, radiation safety, criticality safety, and chemical safety measures.

Margin	For the preclosure period, capability in excess of minimum design requirements in specifications for engineered systems, structures and components. Margin provides for a reserve capability in order to account for uncertainty in the conditions to which the systems, structures, and components will be subjected, to account for variability in the properties of component materials, and to account for the possibilities of overload or understrength due to variations in material dimensions, construction procedures, or simplifications of calculations.
Permanent Closure	The final backfilling of the underground facility, if appropriate, and the sealing of shafts, ramps, and boreholes.
Postclosure	Refers to the period of time after permanent closure of the repository system.
Preclosure	Refers to the period of time before and during permanent closure of the repository system.
Radioactive Decay	The process by which one radionuclide transforms to another nuclide by emission of particles or energy. In some cases, the result is a stable atom, and in other cases, the daughter product is itself radioactive.
Radionuclide	An atom that is radioactive.
Rem	The unit of equivalent radiation dose (the word "rem" is an acronym for Roentgen Equivalent Man). Not all types of radiation produce the same effect in humans. The equivalent dose takes into account the type of radiation and the absorbed dose and its effect. For example, when considering beta, x-ray, and gamma ray radiation, the equivalent dose (expressed in rems) is equal to the absorbed dose (expressed in rads). For alpha radiation, the equivalent dose is assumed to be twenty times the absorbed dose (see also Sievert).
Restricted Area	An area to which access is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials.
Retrieval	The act of intentionally removing radioactive waste from the underground location at which the waste had been previously emplaced for disposal.
Risk	Expected (mean) value of consequences of an undesirable process or event.

Safety Case	The logic, analyses, and calculations that show that the repository system would meet performance objectives.
Safety Margin	The difference between expected performance and the regulatory limit for that performance.
Seismic	Pertaining to, characteristic of, or produced by earthquakes or earth vibrations.
Sievert (Sv)	The standard unit of equivalent dose from ionizing radiation that is applicable to the human body. Often expressed as millionths of a Sievert (micro-sieverts) or thousandths of a Sievert (millisieverts). One Sievert is equivalent to 100 rem; one millisievert (0.001 Sv) equals 0.1 rem.
Source Term	Types and amounts of radionuclides that are the source of a potential release from the repository.
Spent Nuclear Fuel	Fuel that has been withdrawn from a nuclear reactor following irradiation.
Total Effective Dose Equivalent (TEDE)	The sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).
Two-block	A "two-blocking" event occurs during crane hoisting when the load block comes into contact with the upper head block resulting in excessive loads on the wire rope, leading to potential rope failure and drop of the load.
Underground Facility	The underground structure, backfill materials, if any, and openings that penetrate the underground structure (e.g., ramps, shafts, and boreholes, including their seals).
Unrestricted Area	An area, access to which is neither limited nor controlled by the licensee.
Waste Form	The radioactive waste materials and any encapsulating or stabilizing matrix.
Waste Package	The waste form and any containers shielding, packing, and other absorbent materials immediately surrounding an individual waste container. The waste package begins its existence when the outer lid welds are complete and accepted.
Zircaloy	A zirconium alloy used for spent nuclear fuel cladding.

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QA: QA

Civilian Radioactive Waste Management System
Management & Operating Contractor

Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations

Volume II: Postclosure Safety Strategy

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Civilian Radioactive Waste Management System
Management & Operating Contractor

Repository Safety Strategy
Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations

Volume II: Postclosure Safety Strategy

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EXECUTIVE SUMMARY
POSTCLOSURE SAFETY STRATEGY

A key issue for the site recommendation and licensing considerations for a high-level waste repository at Yucca Mountain is the ability of that site to protect the public from any undue radiological risk before and after permanent closure. The U.S. Department of Energy is therefore preparing a case regarding preclosure and postclosure safety of a potential repository system at Yucca Mountain. This document reports the current status of this safety case and the plans to complete it.

This document presents Revision 4 of the Repository Safety Strategy and reflects project developments and information learned since previous revisions. Volume I presents the preclosure safety strategy. This volume, Volume II, presents the postclosure safety strategy. The postclosure safety strategy addresses postclosure performance of a potential repository system at Yucca Mountain, the principal factors of postclosure safety, the current postclosure safety case, and the plans to complete that case to support the site recommendation and licensing considerations.

Postclosure Performance

Yucca Mountain provides a physically and chemically stable rock well below the surface. As a consequence, waste can be emplaced deep underground and isolated from the surface. In the environments underground, temperature, humidity, and chemistry change slowly and permit waste packages and drip shields to maintain their integrity for tens of thousands of years. The waste packages alone are predicted to prevent any release of radionuclides for more than 10,000 years. The drip shield and drift invert provide additional barriers to limit release of radionuclides even if waste packages fail prematurely.

A repository system at the Yucca Mountain site would, however, provide more than stable environments for the engineered barriers. It would include multiple natural and engineered barriers that could provide safety margin and defense-in-depth. The site is arid, and the combination of processes at the surface and the characteristics of the mountain limit the amount of seepage that can enter the emplacement drifts. Chemical conditions limit the concentrations of radionuclides in the water, and the combination of low seepage and these concentration limits constrains the amount of radionuclides that can be released from the engineered barriers even if they are breached.

The repository system also includes barriers to radionuclide migration away from the repository. The unsaturated zone at this site is sufficiently thick that the waste can be emplaced not only more than 200 meters below the surface, but hundreds of meters above the water table. In addition, the site is located more than 20 km from populated areas. Most radionuclides are so immobile in the rock at Yucca Mountain that they cannot migrate readily to populated areas. The combination of the limited ability of radionuclides to be mobilized in the engineered barrier system and the delay of radionuclides that travel away from the system limits the possible mean annual dose to levels not significantly greater than those from natural background radiation. Thus, the multiple natural and engineered barriers provide defense-in-depth and combine to restrict mean annual dose below the radiological exposure limit for at least 80,000 years.

Yucca Mountain is stable and has changed little in the last several million years. The probability of events at this site that could disrupt the repository is low. The probability of igneous activity in the next 10,000 years is less than one chance in 1,000. Analyses of this activity show that the mean annual dose arising from radionuclides that might be mobilized and carried to populated areas would be low. Analyses to assess the robustness of the system with respect to inadvertent human intrusion show that in that case as well, the quantity of radionuclides that could be mobilized would be small and the amounts that could reach populated areas are almost negligible.

Principal Factors of the Current Postclosure Safety Case

The principal factors, those factors that determine whether the postclosure performance objective would be met, are listed in Table ES-1. Their roles in the nominal scenario (the scenario for nominally expected conditions) and the igneous activity scenario are also shown in this table.

Table ES-1. Principal Factors - Nominal and Igneous Scenarios

Principal Factors	Nominal Scenario	Igneous Activity Scenario
Seepage into emplacement drifts	Defense-in-Depth	Contribution to Performance
Performance of the drip shield/drift invert system	Defense-in-Depth	
Performance of the waste package	Contribution to Performance	
Radionuclide concentration limits in water	Defense-in-Depth	Contribution to Performance
Radionuclide delay through the unsaturated zone	Defense-in-Depth	Contribution to Performance
Radionuclide delay through the saturated zone	Defense-in-Depth	Contribution to Performance
Probability of igneous intrusion		Contribution to Performance
Repository response to igneous intrusion		Contribution to Performance

Current Postclosure Safety Case

The focus of the postclosure safety case is the postclosure performance objective to be established by the NRC. In its proposed rule defining this performance objective, the NRC requires that demonstration that this performance objective is met include a performance assessment. To enhance confidence the performance objective is met, however, the DOE considers it prudent to provide more than performance assessment in its postclosure safety case. The postclosure safety case therefore includes the following five elements:

- Performance assessment
- Safety margin and defense-in-depth
- Explicit consideration of potentially disruptive events
- Insights from natural analogues
- Performance confirmation.

Performance assessment provides quantitative estimates of mean annual dose for comparison with the radiological exposure limit of the postclosure performance objective. The analyses consider all credible favorable and potentially adverse features, events, and processes. The principal factors represent the most important of these and the analyses considering them result in mean annual doses that meet the radiological exposure limit of the postclosure performance objective.

Safety margin means the system is designed to perform better than regulatory requirements. The greater the safety margin, the greater the confidence those requirements will be met. Defense-in-depth means utilization of multiple natural and engineered barriers to ensure system performance does not rely unduly on any single barrier. The current postclosure safety case provides substantial safety margin and adequate defense-in-depth.

Explicit consideration of potentially disruptive processes and events means that these processes and events are considered directly and those with a probability of occurrence in the next 10,000 years of greater than one chance in 10,000 are incorporated into the quantitative performance assessment. The current postclosure safety case shows the risk associated with potential igneous activity and inadvertent human intrusion is well below the proposed radiological exposure limit. The case also shows that the probability for significant effects from other potentially disruptive processes or events is less than one chance in 10,000 in 10,000 years.

Insights from natural analogues refer to data from other sites that bear on long-term performance of a repository system at the Yucca Mountain site. In particular, the postclosure safety case includes information that can be obtained relevant to long-term (e.g., millennia) or large-scale (e.g., kilometers) behavior not possible in laboratory and field studies. The current postclosure safety case documents 19 natural analogues of processes important to postclosure performance of a potential repository at Yucca Mountain.

Performance confirmation is the program of testing and analyses that began during site characterization and extends until permanent closure of the repository. This program will confirm estimates of system and barrier performance. A preliminary performance confirmation plan has been developed, but the final plan must wait for finalization of the regulations defining the regulatory requirements for performance confirmation.

Completing the Postclosure Safety Case for Site Recommendation and Licensing Considerations

The postclosure safety case is nearly complete. A few issues remain but they are understood and general approach to address them has been identified. The work needed to address these remaining issues is shown in Table ES-2.

Table ES-2. Work Needed to Address Remaining Issues in the Postclosure Safety Case

Element of Postclosure Safety Case	Issues Remaining in the Postclosure Safety Case	Work Needed to Address Remaining Issues
Performance Assessment	• Management of uncertainties	• Complete the review of treatment of uncertainty and conservatism in TSPA models
	• Reliance on waste package performance	• Conduct additional testing and analysis of waste package degradation
	• Repository-generated heat and coupled effects	• Assess coupled effects • Refine models and conduct performance assessment to assess both above-boiling and lower-temperature repository designs
Safety Margin and Defense-in-Depth	• Role of natural barriers • Reliance on waste package performance	• Evaluate potential for drip shield/drift invert system to provide increased defense-in-depth • Evaluate other process models to determine potential for increased margin and defense-in-depth – Seepage – Emplacement drift environments – In-package transport and transport away from waste package – Drift shadow – Unsaturated and saturated zone transport
Potentially Disruptive Events	• Potential for igneous activity at Yucca Mountain	• Provide conditional results for igneous activity groundwater release scenario
Natural Analogues	• Management of uncertainties	• Evaluate value of studies of natural analogues after license application
Performance Confirmation	• Performance confirmation program	• Complete Performance Confirmation Plan for a License Application

CONTENTS

Page

EXECUTIVE SUMMARY – POSTCLOSURE SAFETY STRATEGY v

ACRONYMS xvii

1 INTRODUCTION 1-1
- 1.1 OBJECTIVE AND SCOPE 1-1
- 1.2 APPLICABILITY OF QUALITY ASSURANCE CONTROLS 1-1
- 1.3 CONCEPTS IMPORTANT TO THE POSTCLOSURE SAFETY STRATEGY 1-1
  - 1.3.1 Postclosure Performance Objective and Waste Isolation 1-1
  - 1.3.2 Risk-Informed, Performance-Based Approach 1-2
  - 1.3.3 Performance Assessment 1-2
  - 1.3.4 Postclosure Safety Case 1-3
  - 1.3.5 Treatment of Uncertainty in the Postclosure Safety Case 1-5
  - 1.3.6 Postclosure Safety Strategy 1-5
- 1.4 ORGANIZATION OF VOLUME II 1-6

2. THE POTENTIAL POSTCLOSURE REPOSITORY SYSTEM 2-1
- 2.1 KEY SITE CHARACTERISTICS 2-1
- 2.2 WASTE ISOLATION ATTRIBUTES 2-1
  - 2.2.1 Limited Water Entering Emplacement Drifts 2-2
  - 2.2.2 Long-Lived Waste Package and Drip Shield 2-2
  - 2.2.3 Limited Release of Radionuclides from the Engineered Barriers 2-3
  - 2.2.4 Delay and Dilution of Radionuclide Concentrations by the Natural Barriers 2-3
  - 2.2.5 Low Mean Annual Dose Even Considering Potentially Disruptive Events 2-3
- 2.3 BARRIERS IMPORTANT TO WASTE ISOLATION 2-4

3. POSTCLOSURE SAFETY ANALYSES 3-1
- 3.1 TOTAL SYSTEM PERFORMANCE ASSESSMENT ANALYSES 3-1
  - 3.1.1 Total System Performance Assessment Estimates for the Nominal Scenario 3-2
  - 3.1.2 Total System Performance Assessment Estimates for Igneous Activity Scenarios 3-4
  - 3.1.3 Total System Performance Assessment for the Inadvertent Human Intrusion Scenario 3-5
- 3.2 DEGRADED BARRIER ANALYSES 3-6
- 3.3 NON-MECHANISTIC JUVENILE FAILURE SCENARIO ANALYSES 3-15
- 3.4 NEUTRALIZED BARRIER ANALYSES 3-21
  - 3.4.1 Neutralization Analyses for Natural Barriers 3-21
  - 3.4.2 Neutralization Analyses for Engineered Barriers 3-23
  - 3.4.3 Other Neutralization Analyses 3-27

CONTENTS (Continued)

Page

4. PRINCIPAL FACTORS OF THE POSTCLOSURE SAFETY CASE 4-1
4.1 IDENTIFYING THE PRINCIPAL FACTORS 4-1
4.1.1 Considerations in Identifying Principal Factors 4-1
4.1.2 Nominal Scenario 4-2
4.1.3 Igneous Activity Groundwater Release Scenario 4-6
4.1.4 Other Scenarios 4-7
4.2 OTHER FACTORS FOR SITE RECOMMENDATION CONSIDERATIONS 4-8
4.2.1 Climate, Infiltration, and Unsaturated Zone Flow 4-8
4.2.2 Effects of Heat on the Flow 4-9
4.2.3 Environments in the Emplacement Drifts 4-9
4.2.4 Environments in the Waste Packages 4-9
4.2.5 Cladding Performance 4-10
4.2.6 Waste Form Performance 4-10
4.2.7 Saturated Zone Flow 4-11
4.2.8 Effect of Wellhead on Radionuclide Concentrations 4-11
4.2.9 Biosphere Transport and Uptake (Biosphere Dose Conversion Factors) 4-11
4.3 PRINCIPAL FACTORS AND WASTE ISOLATION 4-11

5. STATUS OF THE CURRENT POSTCLOSURE SAFETY CASE 5-1
5.1 PERFORMANCE ASSESSMENT 5-1
5.2 SAFETY MARGIN AND DEFENSE-IN-DEPTH 5-2
5.2.1 Safety Margin 5-3
5.2.2 Defense-in-Depth 5-3
5.3 POTENTIALLY DISRUPTIVE EVENTS 5-4
5.3.1 Igneous Activity 5-5
5.3.2 Seismic Activity 5-5
5.3.3 Future Climate Change 5-6
5.3.4 Water Table Rise 5-6
5.3.5 Repository-Generated Disruptions 5-7
5.3.6 Inadvertent Human Intrusion 5-7
5.4 NATURAL ANALOGUES 5-7
5.4.1 Natural Analogue for the Unsaturated Zone Flow System 5-8
5.4.2 Natural Analogues for Deposition of Calcite at Yucca Mountain 5-8
5.4.3 Natural Analogues for Seepage in Emplacement Drifts 5-9
5.4.4 Natural Analogues for Thermal-Chemical-Hydrologic Effects 5-9
5.4.5 Natural Analogue for Radionuclide Delay through the Unsaturated Zone 5-10
5.4.6 Natural Analogues for Radionuclide Delay through the Saturated Zone 5-10
5.4.7 Natural Analogue for Performance of the Waste Package and the Drip Shield 5-12
5.4.8 Natural Analogue for Waste Form Degradation 5-12
5.4.9 Natural Analogue for Igneous Activity Eruptive Processes 5-12
5.5 PERFORMANCE CONFIRMATION 5-13

CONTENTS (Continued)

Page

6. UNCERTAINTY IN THE POSTCLOSURE SAFETY CASE 6-1
   6.1 TREATMENT OF UNCERTAINTY IN THE CURRENT POSTCLOSURE SAFETY CASE 6-1
      6.1.1 Robustness of the System Concept 6-1
      6.1.2 Quantification of Uncertainties 6-2
   6.2 ISSUES REQUIRING FURTHER ATTENTION 6-3
      6.2.1 Role of the Natural Barriers 6-3
      6.2.2 Reliance on Waste Package Performance 6-4
      6.2.3 Repository-Generated Heat and Coupled Effects 6-6
      6.2.4 Potential for Igneous Activity 6-7
      6.2.5 Repository Performance Beyond the Regulatory Period 6-7

7. COMPLETING THE POSTCLOSURE SAFETY CASE FOR SITE RECOMMENDATION AND LICENSING CONSIDERATIONS 7-1
   7.1 PLANS FOR PERFORMANCE ASSESSMENT 7-2
      7.1.1 Complete the Review of the Treatment of Uncertainty and Conservatism in TSPA 7-2
      7.1.2 Conduct Additional Testing and Analysis of Waste Package Degradation 7-2
      7.1.3 Assess Coupled Effects 7-3
      7.1.4 Refine Models and Assess Lower-Temperature Repository Designs 7-3
   7.2 PLANS FOR SAFETY MARGIN AND DEFENSE-IN-DEPTH 7-3
   7.3 PLANS FOR POTENTIALLY DISRUPTIVE EVENTS 7-5
   7.4 PLANS FOR NATURAL ANALOGUES 7-5
   7.5 PLANS FOR PERFORMANCE CONFIRMATION 7-5

8. CONCLUSION AND PROSPECTUS 8-1

9. REFERENCES 9-1
   9.1 DOCUMENTS CITED 9-1
   9.2 CODES, STANDARDS, AND REGULATIONS 9-2

APPENDIX A – POSTCLOSURE SAFETY STRATEGY GLOSSARY A-1

APPENDIX B – REGULATORY FRAMEWORK CONSIDERED FOR THE POSTCLOSURE SAFETY STRATEGY B-1

APPENDIX C – METHODOLOGY FOR ANALYSES USED IN THE POSTCLOSURE SAFETY CASE C-1

APPENDIX D – GENERAL APPROACH TO ENHANCEMENT OF CONFIDENCE IN POSTCLOSURE SAFETY D-1

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FIGURES

Page

1-1. Role of the Postclosure Safety Strategy in Updating the Postclosure Safety Case 1-6

3-1. TSPA Estimate of Repository System Performance for the Nominal Scenario 3-2
3-2. Contribution of Natural and Engineered Barriers to System Performance 3-3
3-3. TSPA Analysis of Igneous Activity Scenarios 3-4
3-4. Nominal Scenario—Degraded Performance of the Unsaturated Zone (UZ) Flow and Seepage Barrier 3-7
3-5. Nominal Scenario—Degraded Performance of the Drip Shield 3-7
3-6. Nominal Scenario—Degraded Performance of the Waste Package 3-8
3-7. Nominal Scenario—Degraded Performance of the CSNF Cladding 3-8
3-8. Nominal Scenario—Sensitivity to Radionuclide Concentrations Limits in Water 3-9
3-9. Nominal Scenario—Degraded Performance of the Unsaturated Zone (UZ) Radionuclide Transport Barrier 3-9
3-10. Nominal Scenario—Degraded Performance of the Saturated Zone (SZ) Radionuclide Transport Barrier 3-10
3-11. Nominal Scenario—Sensitivity to Annual Water Usage Volume 3-10
3-12. Nominal Scenario—Sensitivity to Biosphere Dose Conversion Factors (BDCFs) 3-11
3-13. Nominal Scenario—Sensitivity to Temperatures in the Emplacement Drifts (Backfill vs. No Backfill Design) 3-11
3-14. Igneous Intrusion Groundwater Release Scenario—Sensitivity to Probability of Igneous Intrusion 3-12
3-15. Igneous Intrusion Groundwater Release Scenario—Sensitivity to Effects of Igneous Intrusion on Waste Packages and Drip Shields 3-12
3-16. Non-Mechanistic Juvenile Waste Package Failure Scenario Base Case Results 3-15
3-17. Non-Mechanistic Juvenile Failure Scenario—Degraded Performance of the Unsaturated Zone (UZ) Flow and Seepage Barrier 3-17
3-18. Non-Mechanistic Juvenile Failure Scenario—Degraded Performance of the Drip Shield 3-17
3-19. Non-Mechanistic Juvenile Failure Scenario—Degraded Performance of CSNF Cladding 3-18
3-20. Non-Mechanistic Juvenile Failure Scenario—Sensitivity to Radionuclide Concentration Limits 3-18
3-21. Non-Mechanistic Juvenile Failure Scenario—Degraded Performance of the Unsaturated Zone (UZ) Radionuclide Transport Barrier 3-19
3-22. Non-Mechanistic Juvenile Failure Scenario—Degraded Performance of the Saturated Zone (SZ) Radionuclide Transport Barrier 3-19
3-23. Neutralized Performance of the Unsaturated Zone (UZ) Flow and Seepage Barrier 3-22
3-24. Neutralized Performance of Unsaturated Zone (UZ) and Saturated Zone (SZ) Radionuclide Transport Barriers 3-23
3-25. Neutralized Performance of Drip Shield 3-24
3-26. Neutralized Performance of Waste Package (WP) 3-24

FIGURES (Continued)

Page

3-27. Neutralized Performance of Drip Shield (DS) and Waste Package (WP) Together 3-25
3-28. Neutralized Performance of Drift Invert Radionuclide Transport Barrier 3-25
3-29. Neutralized Performance of CSNF Cladding 3-27
3-30. Neutralization Analysis of Radionuclide Concentration Limits 3-28

TABLES

Page

4-1. Principal Factors of the Postclosure Safety Case....4-2
4-2. Principal Factors—Nominal and Igneous Activity Scenarios....4-3

7-1. Plans for Work to Complete the Postclosure Safety Case for Site Recommendation and Licensing Considerations....7-1

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ACRONYMS

BDCF	biosphere dose conversion factor
CFR	Code of Federal Regulations
CSNF	commercial spent nuclear fuel
DOE	U.S. Department of Energy
DSNF	DOE spent nuclear fuel
EPA	U.S. Environmental Protection Agency
FEP	feature, event, and/or process
FR	Federal Register
HLW	high-level waste
NRC	U.S. Nuclear Regulatory Commission
PMR	process model report
RSS	Repository Safety Strategy
RSS4	Repository Safety Strategy Revision 4
SCP	Site Characterization Plan
SZ	saturated zone
TSPA	total system performance assessment
UZ	unsaturated zone
WP	waste package

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1. INTRODUCTION

1.1 OBJECTIVE AND SCOPE

A key issue for the site recommendation and license application considerations for a high-level waste repository at Yucca Mountain is the ability of that site to protect the public from any undue radiological risk before and after permanent closure. The U.S. Department of Energy (DOE) is therefore preparing a case regarding preclosure and postclosure safety of a potential repository system at Yucca Mountain. This document, Revision 4 of the Repository Safety Strategy (RSS), reports the current status of this safety case and the plans to complete it.

This document updates its predecessors to reflect project developments and information learned since previous revisions. This revision is based on the work to date, including RSS workshops conducted from February through June 2000. While previous revisions focused solely on postclosure safety, Revision 4 also addresses the preclosure safety case. Volume I presents the preclosure safety strategy. This volume, Volume II, presents the postclosure safety strategy.

1.2 APPLICABILITY OF QUALITY ASSURANCE CONTROLS

Revision 4 of the RSS is subject to controls specified in the *Quality Assurance Requirements and Description* (DOE 2000). Revision 4 has been planned, developed, checked, reviewed, and approved in accordance with applicable quality assurance procedures.

This revision of the RSS may be affected by technical product input information that requires confirmation. Any changes to the document that may occur as a result of completing the confirmation activities will be reflected in subsequent revisions. The status of the input information quality may be confirmed by review of the Document Input Reference System database.

1.3 CONCEPTS IMPORTANT TO THE POSTCLOSURE SAFETY STRATEGY

1.3.1 Postclosure Performance Objective and Waste Isolation

This strategy focuses on the postclosure performance objective for the repository system. This performance objective is the following:

> Provide multiple natural and engineered barriers capable of inhibiting movement of radionuclides to populated areas so that regulatory radiological exposure limits that apply for 10,000 years can be met.

The term, "waste isolation," is used in this strategy to delineate the capability of the repository system to meet the postclosure performance objective. In the context of a repository system, waste isolation encompasses the concept that the waste can be emplaced deep enough and far enough from the public that the members of the public can be adequately protected from radiological exposures. The regulatory framework defining this performance objective and the approach required to meet it are summarized in Appendix B.

Section 2 describes the potential postclosure repository system and the attributes that would contribute to waste isolation at Yucca Mountain. Section 2 also identifies the natural and engineered barriers that could be important to waste isolation. The principal factors of waste isolation, that is, those factors that determine if the postclosure performance objective is met, are described in Section 4.

1.3.2 Risk-Informed, Performance-Based Approach

The U.S. Nuclear Regulatory Commission (NRC) has been moving to a more risk-informed regulatory framework in which insights from risk assessments are used to focus on the areas of greatest risk to the public. The NRC's proposed regulation for a repository system at Yucca Mountain formalizes this approach. The proposed regulation is performance-based in that it specifies performance objectives that must be met. It is risk-informed in that decision-making takes into account the results and findings from a quantitative risk assessment as well as other factors.

The risk-informed, performance-based approach contrasts with a risk-based approach. In a risk-based approach, decision-making would be based solely on the results of a quantitative risk assessment. That is, the results of the risk assessment would be used to compare with numerical radiological exposure limits and these results would be the only basis for determining whether the postclosure performance objective is met. This approach would require complete confidence in the risk assessment methodology and models. Although the risk-informed approach also requires confidence in the risk assessment, it does not demand perfect knowledge. The risk-informed approach permits consideration of

- A broader set of concerns than might be accounted for in the risk assessment
- Rational prioritization of these concerns in terms of both risk estimates and additional judgment
- A broader set of measures to address the concerns in addition to development of the risk assessment methodology.

Application of the risk-informed, performance-based approach to the considerations for a repository system at Yucca Mountain is discussed in Appendix B.

1.3.3 Performance Assessment

Performance assessment of a repository system takes into account all the factors that determine postclosure performance of the natural and engineered barriers of that system in order to estimate the annual dose to an individual in the critical group. The estimate includes the results for various scenarios, including scenarios for potentially disruptive processes and events. The annual doses for the scenarios are each weighted by the probability of occurrence of their respective scenarios and summed to provide an overall estimate, or mean annual dose, for the repository system.

This mean annual dose calculated in the performance assessment provides the measure of risk to support NRC's risk-informed, performance-based regulatory philosophy. In particular, NRC's

proposed regulation specifies that this measure is to be considered in evaluating the postclosure performance objective. That is, the proposed postclosure performance objective includes a radiological exposure limit in terms of annual dose (total effective dose equivalent) that applies to the mean, or probability-weighted, annual dose. In the context of the risk-informed approach, the performance assessment is viewed as providing the risk estimates to be considered in determining whether this postclosure performance objective is met.

Appendix B discusses the role of performance assessment in the current regulatory framework and the approach to performance assessment in the postclosure safety case is summarized in Appendix C. Results of performance assessment analyses considered in the current postclosure safety case are provided in Section 3.

1.3.4 Postclosure Safety Case

The postclosure safety case is an assessment of long-term safety of the repository system prepared to support each stage of decision-making. This case includes the results of performance assessment, other findings, and a statement of confidence in these findings. The postclosure safety case is developed in accordance with the postclosure safety strategy and includes information regarding the natural and engineered barriers developed following that strategy. The case acknowledges uncertainties in the assessment and the existence of any unresolved issues.

The current status of the postclosure safety case is summarized in Section 5. It is being developed to support site recommendation and licensing considerations. This postclosure safety case focuses on the postclosure performance objective of the regulations that apply to these considerations. At the same time, the postclosure safety case is being developed to serve the broader purpose of providing confidence to all stakeholders including the public and oversight groups, as well as regulatory agencies.

For all of these purposes, the risk-informed, performance-based approach used for the postclosure safety case utilizes risk insights, other analyses, and judgment, including safety margin and defense-in-depth (NRC 1999). That is, because the approach is risk-informed, rather than risk-based, the postclosure safety case provides information in multiple areas to address not only the postclosure performance objective of the applicable regulations, but also the broader issue of confidence building. The case therefore has the following elements:

- Performance assessment
- Safety margin and defense-in-depth
- Explicit consideration of potentially disruptive events
- Insights from natural analogues
- Performance confirmation.

Performance assessment provides the quantitative estimates of expected annual dose for comparison with the radiological exposure limit of the postclosure performance objective. In addition, performance assessment includes analyses to determine the capability of individual natural and engineered barriers to isolate the waste. The performance assessment is used to identify the principal factors of the postclosure safety case, those factors that determine

postclosure performance. The principal factors are identified to increase transparency of the postclosure safety case and to specify the areas where uncertainty is important.

Safety margin means the system is designed to perform better than regulatory requirements. The greater the safety margin, the greater the confidence those requirements will be met. Defense-in-depth means utilization of multiple natural and engineered barriers to ensure system performance does not rely unduly on any single barrier. The postclosure safety case describes the measures taken to ensure safety margin and defense-in-depth and includes analyses of the degree of margin and defense-in-depth that would be provided by the repository system.

Explicit consideration of potentially disruptive events means that these events are addressed both directly those whose probability of occurrence in the next 10,000 years is greater than one chance in 10,000 are incorporated in the quantitative performance assessment. The current postclosure safety case includes potential igneous activity and inadvertent human intrusion. In addition to this quantitative information, the postclosure safety case includes qualitative discussion of all features, events, and processes (FEPs) with potential for disrupting the repository to ensure that information important to the safety assessment has been duly considered.

Insights from natural analogues refer to data from other sites that bear on long-term performance of a repository system at the Yucca Mountain site. In particular, the postclosure safety case includes information that can be obtained to evaluate long-term (e.g., millennia) or large-scale (e.g., kilometers) behavior not possible in laboratory and field studies.

Performance confirmation is the program of testing and analyses beginning during site characterization and extending to permanent closure of the repository to confirm estimates of system and barrier performance. The postclosure safety case provides a plan for testing and analysis as part of this program to evaluate performance of the system, monitor whether conditions are as anticipated, and confirm barriers are performing as expected.

These five elements have many common ties and are, therefore, not completely independent. They rely on common conceptual models for future behavior of the system and the way FEPs combine to isolate waste. Analyses conducted in several of these areas use the same performance assessment models and are therefore subject to common sources of uncertainty. Nevertheless, they do provide information in different areas, which increases confidence in the safety case. Analyses of the degree of safety margin and defense-in-depth use the same performance assessment models used to estimate the mean annual dose; however, these analyses evaluate additional factors to provide an assessment of the robustness of that estimate. Likewise, potentially disruptive processes and events are addressed within the quantitative performance assessment; examining them individually in addition to incorporating them into the estimate of overall risk provides information in addition to the risk estimate. In this way, the postclosure safety case provides information to focus attention on the principal factors determining postclosure safety, to understand the most important sources of uncertainty, and to support decision-makers and other interested parties in their efforts to assess postclosure safety of the system.

1.3.5 Treatment of Uncertainty in the Postclosure Safety Case

Critical to the postclosure safety case is the treatment of uncertainty. Uncertainties in the available scientific understanding, data, and models for the factors that determine barrier performance are inevitable because of system complexity and the long time scales involved. Many of these uncertainties are explicitly taken into account in the performance assessment. Alternate conceptual models and probability distributions to represent uncertainty and variability in model parameters are incorporated into the performance assessment so that variance around the mean of the annual dose can be considered in evaluating whether the performance objective is met.

At the same time, it is prudent to assume that not all uncertainty is represented explicitly in the performance assessment. Some uncertainties arise from complexities that prevent complete mechanistic understanding, validation of models, or practicable analysis. Other uncertainties are associated with unanticipated or unknown effects, such as failure mechanisms not yet revealed in spite of extensive testing. The postclosure safety case therefore employs additional measures consistent with the risk-informed, performance-based regulatory framework. The aim of these measures is to provide a safety case in which, while acknowledging the presence of uncertainty, there is sufficient confidence in the postclosure safety that decisions by the DOE and the NRC to proceed to the next stage of repository development can be made. These measures include those that aim at increasing

- Robustness of the system
- Quantification of uncertainties.

Both of these approaches are addressed within the postclosure safety case. Robustness in the system concept is addressed through adoption of multiple engineered and natural barriers to provide safety margin and defense-in-depth and in providing information in a number of different areas to increase confidence in the postclosure safety case. Quantification of uncertainties is addressed through the technical basis for the performance assessment models, comprehensive accounting of uncertainties within the performance assessment, the use of appropriate conservatism to address uncertainties, and model validation activities.

Appendix D provides a summary of DOE's general approach to the treatment of uncertainty. Section 6 summarizes the treatment of uncertainty in the current postclosure safety case and identifies the key issues that remain to be addressed to complete the postclosure safety case for site recommendation and licensing considerations.

1.3.6 Postclosure Safety Strategy

The postclosure safety strategy documents the current postclosure safety case and provides the plans to complete that case for site recommendation and licensing considerations. The general picture of the strategy and its relationship to the postclosure safety case is shown in Figure 1-1.

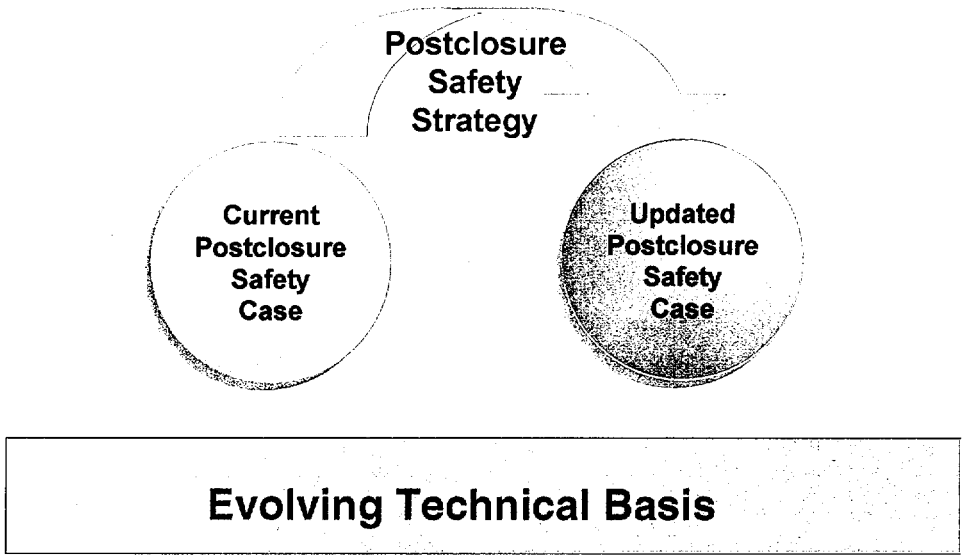

Figure 1-1. Role of the Postclosure Safety Strategy in Updating the Postclosure Safety Case

This figure indicates how the postclosure safety case and its supporting technical basis evolve in response to the postclosure safety strategy. As plans for development of the postclosure safety case are followed, site information is acquired, design enhancements are evaluated, and performance assessment models are developed. The new information is used to update the postclosure safety case.

Revision 4 of the strategy takes into account improvements in process models and changes in the design that are explained later in this document. It finalizes the set of principal factors that were identified on a preliminary basis in Revision 3. Other than principal factors associated with potentially disruptive events (which were not evaluated in Revision 3), the final set is not significantly different from the preliminary set, reflecting convergence in the understanding of the principal factors. More generally, this revision reflects maturation in the postclosure safety case and suggests that future changes for the site recommendation and licensing considerations are not likely to be substantial. The main thrust of future work would be to provide additional justification of performance assessment models. Wholesale changes in the postclosure safety case from this point are therefore not expected.

1.4 ORGANIZATION OF VOLUME II

Sections 1 through 4 of this volume introduce the postclosure safety strategy. Section 1 provides the overview of the strategy. Section 2 provides a brief description of the potential repository system, focusing on those site characteristics, attributes, and natural and engineered barriers potentially important to waste isolation. Section 3 presents postclosure performance assessment analyses for this potential repository system. Section 4 summarizes the principal factors emerging from these analyses.

Section 5 gives the status of the current postclosure safety case. This section summarizes the performance assessments and the other elements of the postclosure safety case: safety margin and defense-in-depth, disruptive processes and events, natural analogues, and performance confirmation.

Section 6 discusses the treatment of uncertainties in the current postclosure safety case. It discusses the efforts to address these uncertainties through measures to enhance robustness of the repository system concept. It also discusses the efforts to identify and quantify the uncertainties and incorporate them into the performance assessment. Section 6 summarizes key issues of particular importance that would need to be addressed to complete the postclosure safety case for site recommendation and licensing considerations. Section 7 presents the plans to address these issues and to complete the postclosure safety case for these considerations. Section 8 summarizes the entire postclosure safety strategy and Section 9 provides the references used in Volume II.

This volume also includes appendices with supplementary information. Appendix A provides a glossary of terms used in Volume II. Appendix B gives the regulatory framework pertaining to postclosure safety for this revision of the strategy. Appendix C summarizes the methodology for the analyses conducted for this strategy. Appendix D provides an overview of the general approach to enhancing confidence in postclosure safety and the management of uncertainty.

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2. THE POTENTIAL POSTCLOSURE REPOSITORY SYSTEM

2.1 KEY SITE CHARACTERISTICS

Yucca Mountain is located approximately 160 km (100 miles) northwest of Las Vegas, Nevada, and approximately 20 km north of Amargosa Valley, the nearest populated area. The mountain is a long, broad ridge that has changed little over the past million years. For these and other reasons, the Yucca Mountain site was chosen for characterization studies to determine its potential for repository development.

The site appeared to offer a number of characteristics favorable to waste isolation. Location of a potential repository at Yucca Mountain deep underground and far from populated areas would serve to separate emplaced waste from populated areas. The available area is large enough to accommodate 70,000 metric tons of commercial spent nuclear fuel (CSNF), high-level waste (HLW), and other nuclear wastes that would be contained in approximately 11,000 waste packages. These waste packages could be emplaced in drifts located more than 200 m below the surface, far enough to isolate them from surface events, such as storms and daily changes in temperature and humidity. The water table is sufficiently deep at this site that the emplacement drifts at this depth would still be separated from the water table by at least 200 m (CRWMS M&O 2000a, Section 9.4.1). Thus, the waste can be isolated from the surface and from groundwater at this site.

The properties of the rock at the candidate repository horizon and the environments (e.g., temperature, relative humidity) that would occur in the repository are understood well enough to identify engineered barrier materials that would be compatible with them. Thus, the properties of the rock would favor long service life of engineered barriers designed to prevent exposure of the wastes to water and to contain the radionuclides. Materials that would be durable in the repository have been identified.

Finally, properties of the rock sorb radionuclides and would, therefore, delay migration of radionuclides that might escape from the engineered barrier system. These properties are such that many of the radionuclides in the waste are immobile at Yucca Mountain. The combination of these characteristics and the other characteristics of the site are being considered in the evaluations of the suitability of the Yucca Mountain site for repository development.

2.2 WASTE ISOLATION ATTRIBUTES

A central question for these evaluations is whether the characteristics of the Yucca Mountain site would support a repository system that could meet the postclosure performance objective. This postclosure performance objective focuses on the ability of the natural and engineered barriers of this system to prevent transport of radionuclides to populated areas such as Amargosa Valley. Consequently, the question becomes whether these characteristics provide attributes compatible with performance of these barriers.

The following attributes[1] of the system would lend themselves to this objective:

- Limited water entering emplacement drifts
- Long-lived waste package and drip shield
- Limited release of radionuclides from the engineered barriers
- Delay and dilution of radionuclide concentrations provided by the natural barriers
- Low mean annual dose even considering potentially disruptive events.

2.2.1 Limited Water Entering Emplacement Drifts

This attribute is discussed at length in CRWMS M&O (2000b). The Yucca Mountain site is arid, with precipitation averaging less than 200 mm (less than 8 in.) per year (CRWMS M&O 2000b, Section 3.5.2.5). Little of this precipitation percolates into the mountain. Most of it runs off the mountain, evaporates, or is taken up by roots of water-seeking desert plants and transpired back into the atmosphere. The amount of water that actually percolates to depth at Yucca Mountain is too low to be measured directly. Indirect evidence indicates that currently the deep percolation flux averages less than 5 mm (less than 0.25 in.) per year (CRWMS M&O 2000b, Section 3.5.2.5). A larger flux might occur in a future wetter period. Estimates based upon information for past pluvial periods indicate an average percolation flux possibly as high as 33 mm (less than 2 in.) per year (CRWMS M&O 2000b, Section 3.5.2.5). Not all of this water is likely to seep into the emplacement drifts. For example, at the low percolation fluxes indicated for Yucca Mountain, capillary forces tend to hold water in the pores and fractures of the rock and prevent water from seeping into large openings such as the emplacement drifts (CRWMS M&O 2000b, Section 3.9.3.5). The degree to which this attribute supports waste isolation is considered later in this report.

2.2.2 Long-Lived Waste Package and Drip Shield

This attribute is discussed in CRWMS M&O (2000c). The repository design relies on a robust waste package and a drip shield over the waste package, both constructed of materials determined to be corrosion resistant under the full range of environments in the emplacement drifts. The waste package design includes a corrosion-resistant outer barrier of at least 2 cm of nickel-chromium-molybdenum alloy. The current drip shield design includes a single barrier of 1.5 cm of titanium alloy, a strong and long-lasting material different from that used in the waste package. The waste package outer barrier and the drip shield are each expected to last tens of thousands of years in the repository environments (CRWMS M&O 2000c, Section 5), consequently providing a set of redundant barriers capable of preventing seepage from contacting the waste. Again, the specific role of the waste package and drip shield in providing waste isolation is considered later in this report.

[1] This list of key attributes is more comprehensive than that provided in previous versions of this strategy. The list has been expanded to address potentially disruptive events. In addition, the wording of the attributes has changed to reflect emphasis on the role of the drip shield in providing defense-in-depth in preventing exposure of waste to water.

2.2.3 Limited Release of Radionuclides from the Engineered Barriers

Limits to the release of radionuclides from the engineered barrier system are discussed in CRWMS M&O (2000d) and CRWMS M&O (2000e).[2] One of these limits is the waste form, itself, such as, the solid uranium oxide ceramic or HLW glass in which the radionuclides are fixed. Another constraint is the limits to concentrations of radionuclides in water. For example, many of the radionuclides are sufficiently insoluble that they are not amenable to mobilization even if the waste form degrades.

An additional limit on radionuclide release from the engineered barriers is their transport behavior in the waste package and the engineered barriers outside the waste package. For limited flow conditions, transport is also limited by diffusion out of the waste package. This transport would be affected by the waste-generated heat that elevates temperatures and removes moisture. Migration of radionuclides through the drift invert below the waste package would be limited by the transport characteristics of the invert material, including its retardation characteristics and diffusion properties (CRWMS M&O 2000e, Section 3.1.3.4). The degree to which the properties of the system limit release from the engineered barrier system is evaluated in the context of system performance estimates later in this report.

2.2.4 Delay and Dilution of Radionuclide Concentrations by the Natural Barriers

This attribute is discussed in CRWMS M&O (2000b) and CRWMS M&O (2000f). The distance of the repository from the water table provides ample rock (more than 100 m) to sorb, diffuse, and delay movement of radionuclides to the water table. The sorption and matrix diffusion capacity of the rocks above the water table retard migration of the radionuclides (CRWMS M&O 2000b, Sections 3.11.2.4 and 3.11.2.5). Rocks below the water table and the alluvial deposits about 15 km from the repository also have retardation capacity (CRWMS M&O 2000f, Section 3.2.4). Migration of many of the radionuclides would therefore be retarded in the saturated zone below the water table. Accordingly, migration of many of the radionuclides away from the repository would be limited even if released from the waste packages. Those radionuclides that are released into the groundwater disperse as they migrate and the concentrations become more dilute (CRWMS M&O 2000b, Section 3.11.2.3 and CRWMS M&O 2000f, Section 3.2.4.4). The ultimate concentrations are determined by the amount of radionuclides and volume of water taken up in a well and used by individuals.

The role of this attribute in meeting the postclosure performance objective is considered later in this report.

2.2.5 Low Mean Annual Dose Even Considering Potentially Disruptive Events

Many potentially disruptive events would play little role at the Yucca Mountain site because their probability of occurrence is too low to warrant consideration. However, potentially disruptive events that need to be considered include inadvertent human intrusion, igneous

[2] The waste that would be emplaced in the repository contains hundreds of radionuclides. Many of these are short-lived and decay within a few hundred years. Others are relatively insoluble and dissolve to only trace quantities. Considering these factors, fewer than 30 radionuclides present a potential for significant radiological exposure and are included in the current total system performance assessment (CRWMS M&O 2000d, Section 3.1.1.1).

activity, and other processes and events. With respect to the potential for inadvertent human intrusion, location of the repository system at depth limits the potential for this possibility. There is little incentive for exploratory drilling at Yucca Mountain because this site does not provide geologic signatures suggesting energy or mineral reserves (CRWMS M&O 2000a, Section 4.10). However, because drilling cannot be precluded at this or any other site, the possibility of inadvertent human intrusion is being evaluated in the postclosure safety case to assess the robustness of the repository concept in the event of human intrusion. With respect to potential igneous activity, the tuffs composing Yucca Mountain resulted from volcanic igneous activity between 14 and 11 million years ago and activity has diminished in the region since that time. The youngest volcano in the vicinity of the site is the Lathrop Wells volcano, which is more than 74,000 years old (CRWMS M&O 2000g, Section 3.1.1.1).

The effect of these potentially disruptive events on the ability of the repository system to meet the postclosure performance objective is considered later in this report.

2.3 BARRIERS IMPORTANT TO WASTE ISOLATION

The summary of key waste isolation attributes leads to a natural identification of the barriers potentially important to waste isolation. Natural barriers include the following:

- Surficial soils and topography, which limit water infiltration
- Unsaturated rock layers above the repository horizon, which limit water flux in the repository emplacement drifts
- Unsaturated rock layers below the repository horizon, which limit radionuclide transport
- Tuff and alluvial aquifers, which limit radionuclide transport in the saturated zone.

Engineered barriers included the following:

- A drip shield, which limits the water contacting the waste package and the water available for advective transport through the waste package and drift invert
- A waste package, which limits the water contacting the waste form
- CSNF cladding, which limits the water contacting the waste matrix
- A waste form that limits rate of release of radionuclides to the water that contacts the waste
- A drift invert, which limits the rate of release of radionuclides to the natural barriers.

As discussed in Section 2.2.1, near-surface hydrologic processes limit the flux of water at repository depth to a small fraction of the precipitation flux incident on Yucca Mountain. Capillary forces in the rock limit seepage into underground openings because the flux at depth is low.

While the drip shield remains intact, it will divert water from the waste. Further, the drip shield plays a role in limiting transport of radionuclides that might be released from the waste packages. Radionuclides could be released by advection (i.e., transport by flowing water) or by diffusion. The drip shield will prevent flow and therefore preclude advective transport. In addition, because the diffusion coefficient of the drift invert under the waste packages depends upon the moisture conditions, diversion of water by the drip shield will also affect diffusive transport.

The waste package also prevents exposure of the waste within it to water as long as it remains intact. The waste package is corrosion resistant and designed specifically to protect the waste. Further, as long as this barrier remains intact, radionuclides cannot be transported away from the repository. Even after the waste packages are breached, the waste package could affect transport of radionuclides; that is, if the breach is small, the amount of radionuclides that can be released will be small. If the breach is very small (e.g., a hairline fracture), transport of radionuclides through the breach could be negligible.

The invert forming the emplacement drift floor provides a barrier to movement of radionuclides from the waste package to the host rock. The transport properties of the drift invert directly affect advective transport of radionuclides through this barrier. In addition, the diffusion coefficient of the drift invert will affect diffusive transport through it.

Radionuclides released from the engineered barriers would reach the rock units of the unsaturated zone. These rock units, including the Topopah Springs welded, the Calico Hills nonwelded, and the Crater Flat undifferentiated unit, provide barriers to migration of radionuclides. Migration of radionuclides through the solid rock matrix would be relatively slow; however, these units are fractured and more rapid migration could occur through the fractures. The net effectiveness of this barrier therefore depends on the relative distribution of the radionuclides between matrix and fracture migration.

Radionuclides that eventually reach the water table could enter the saturated zone flow system. The saturated zone provides another barrier to radionuclide migration. Directly under the repository, the effectiveness of this barrier depends on the transport characteristics of the volcanic aquifers that dominate the flow system. Further away, near Amargosa Valley, the effectiveness of the barrier is enhanced by the potential for retardation in the valley fill alluvium.

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3. POSTCLOSURE SAFETY ANALYSES

In addition to a sound understanding of the waste isolation attributes of the system, the postclosure safety case includes quantitative analyses of postclosure repository system performance. These analyses are the basis for the identification of the principal factors presented in Section 4. In addition, these analyses are used to evaluate the likelihood of meeting the numerical radiological exposure limit of the postclosure performance objective. They, alone, are insufficient to determine whether the postclosure performance objective is met and therefore do not constitute the entire postclosure safety case. Nevertheless, these analyses are central to that case, which is summarized in Section 5.

These analyses include (1) total system performance assessment (TSPA) analyses, (2) degraded barrier analyses studies, (3) juvenile failure scenario analyses, and (4) neutralized barrier analyses. Results of these analyses relevant to the identification of principal factors and the current postclosure safety case are summarized in the following sections.

3.1 TOTAL SYSTEM PERFORMANCE ASSESSMENT ANALYSES

TSPA analyses provide quantitative estimates of expected performance. The estimates are expressed in terms of the mean annual dose to an individual 20 km from the repository in Amargosa Valley. NRC's proposed regulation specifies a radiological exposure limit that is applicable for 10,000 years after permanent closure (see Appendix B).

The general approach to the estimate of the mean annual dose follows the methodology summarized in Appendix C. The approach takes into account all FEPs that could affect postclosure performance and that have a probability of occurrence in the next 10,000 years greater than one chance in 10,000. These FEPs are taken into account within TSPA component models linked for the computation of the performance of the barriers of the system, migration of radionuclides to Amargosa Valley, and the estimate of the potential dose rate.

The approach is probabilistic. That is, uncertainties in the component models are accounted for in terms of probability distributions for the parameters of these models. The uncertainties are sampled through repeated realizations in a Monte Carlo approach (see Appendix C) to provide a probability distribution for the calculated dose rate. This approach permits not only an estimate of the mean annual dose, but also the variance around the mean arising from parameter uncertainty.

The TSPA analyses presented here provide estimates of repository performance for 100,000 years after permanent closure. Discussion throughout this section compares results with the 10,000-year radiological exposure limit of NRC's proposed rule (see Appendix B). This comparison is intended only to provide perspective and does not reflect an opinion regarding the final standard implemented by the NRC. The discussion considers different scenarios: (1) the nominal scenario (i.e., the scenario for nominally expected repository conditions); (2) igneous activity scenarios; and (3) a scenario for inadvertent human intrusion. Each of these is considered separately.

3.1.1 Total System Performance Assessment Estimates for the Nominal Scenario

Figure 3-1 shows the estimate of the dose rate to a person in Amargosa Valley as a function of time for the nominal scenario. Results are given for 300 realizations. The probability distributions for the FEPs assumed to occur in this scenario are sampled for each realization. This analysis is reported in Section 4.1 of the report of the current TSPA analyses (CRWMS M&O 2000h). The specific data sets and record of the simulation are found in the table in Appendix G of this reference, tracking them with the identification number found above the frame of Figure 3-1.

The mean of the 300 realizations calculated for this scenario is shown in Figure 3-1 as a function of time. To provide additional perspective on the range of performance, the figure also shows the 5th and 95th percentile and the median (50th percentile) dose rate curves. No waste package fails before 10,000 years in any of these realizations (CRWMS M&O 2000h, Section 3.4.3). Because of the long lifetime of the waste package, the mean annual dose is zero for more than 10,000 years and below 25 mrem/year for more than 80,000 years.

The mean annual dose remains below 100 mrem/year for at least 100,000 years. This level is less than the exposure people receive each year from natural background radiation (more than 300 mrem/year) (NCRP 1987, p. 52). The calculated dose rate in the first 60,000 years results primarily from radionuclides that are relatively mobile, mainly technetium-99, iodine-129, and neptunium-237; after 60,000 years, the calculated dose rate is dominated by neptunium-237 and plutonium-239 (CRWMS 2000h, Section 4.1.1). The spread between the 5th and 95th percentile curves indicates the range in the dose rate estimate due to uncertainties taken into account in the current TSPA estimate.

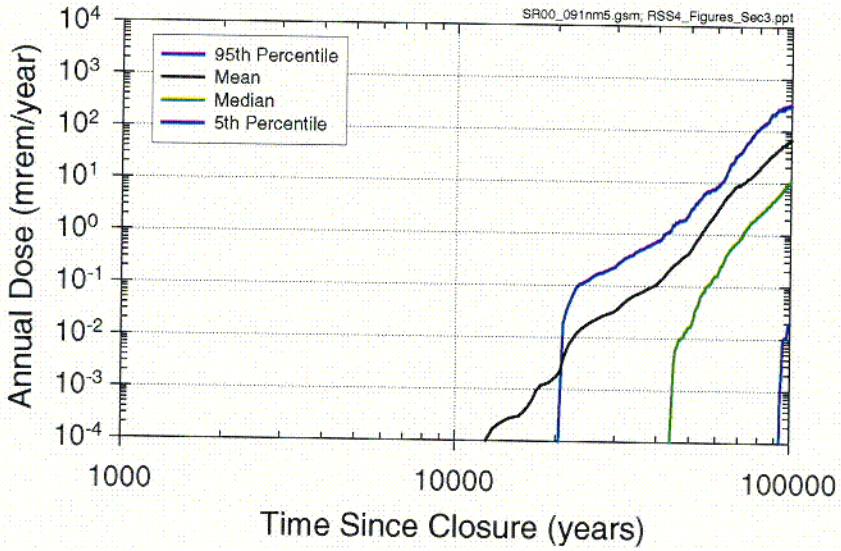

Figure 3-1. TSPA Estimate of Repository System Performance for the Nominal Scenario

Figure 3-2 indicates the role of the natural and engineered barriers in repository system performance. Again the analyses are probabilistic; however, only 100 realizations have been calculated to develop the mean annual dose curves. The upper curve shows the annual dose that would be obtained from the radionuclides considered in these analyses without the benefit of any repository system barriers. This curve is calculated neutralizing all natural and engineered barriers (CRWMS M&O 2000h, Appendix G); the results are equivalent to those estimated by calculating the concentrations of the radionuclides in the repository at their concentration limit and giving this water directly to a person in Amargosa Valley without any dilution, dispersion, or delay, i.e., without the benefit of any repository system barriers.

Figure 3-2 also shows a result indicating the performance of the natural barriers alone. This calculation includes the effect of limited seepage into the emplacement drifts, delay in radionuclide travel to Amargosa Valley through the unsaturated and saturated zones, and dilution in the community water supply. These barriers are effective and reduce the mean annual dose by more than six orders of magnitude. As will be discussed in Section 7, the models for the natural barriers are conservative in several respects and further development of these models might result in an even lower mean annual dose.

Finally, Figure 3-2 shows the result with full contribution of all barriers. In addition to the effects considered previously, this calculation includes the engineered barriers (i.e., the waste package, drip shield, and drift invert) and CSNF cladding. The result shows the improved performance of the system that includes both natural and engineered barriers.

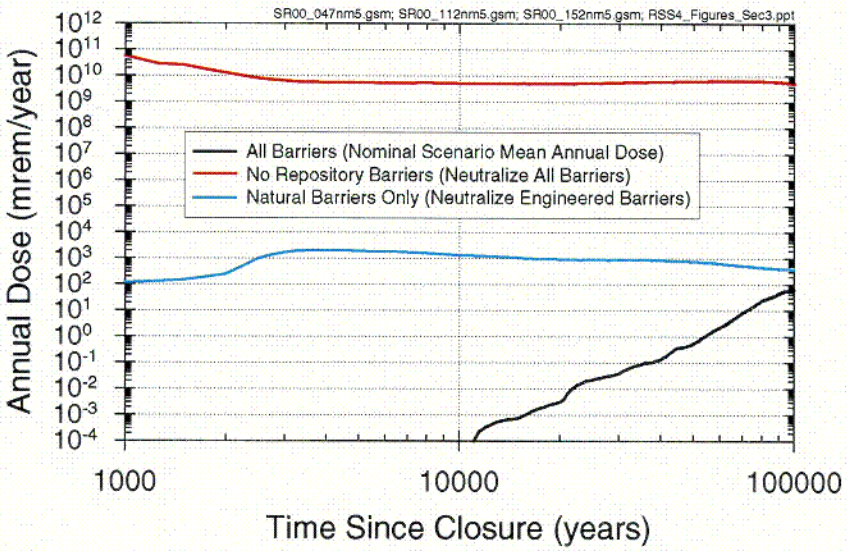

Figure 3-2. Contribution of Natural and Engineered Barriers to System Performance

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3.1.2 Total System Performance Assessment Estimates for Igneous Activity Scenarios

Figure 3-3 shows the results of analyses of potential igneous activity. The situation analyzed is intrusion of magma into the repository from such activity. In some cases, the intrusion would be accompanied by an eruption to the surface; in others, magma intrudes into the emplacement drifts but no eruption occurs. These two possibilities are represented by two different igneous activity scenarios. The first considers eruption of magma through the repository with waste carried to the surface and into the atmosphere by the eruption. Ash and waste particles are dispersed in the atmosphere and some settle in Amargosa Valley. The second scenario considers intrusion of magma into the repository drifts without an associated volcanic eruption. The scenario takes into account damage of waste packages and drip shields by the intruding magma and concomitant exposure of waste to the flow of water through the repository. This flow can mobilize the radionuclides that are then transported down to the water table and out to Amargosa Valley. The conceptual models for both scenarios are described in Section 3.1.1 of *Disruptive Events Process Model Report* (CRWMS M&O 2000g). For this discussion, the first scenario is referred to as the "eruptive release" scenario and the second as the "groundwater release" scenario.

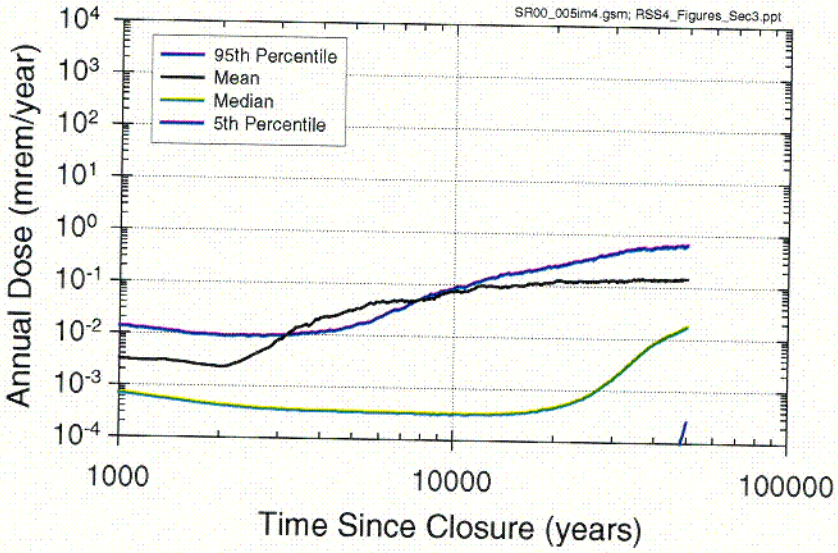

Figure 3-3. TSPA Analysis of Igneous Activity Scenarios

The analysis of these two scenarios is presented in Section 4.2 of the report of the current TSPA analyses (CRWMS M&O 2000h). Figure 3-3 shows the sum of the results for both scenarios. The results for each scenario are calculated for 5,000 different realizations and take into account the probability of occurrence of the scenario. This probability of occurrence in 10,000 years is very low. Using the mean annual frequency for the scenarios (see Tables 3.10-4 and 3.10-5 of CRWMS M&O 2000h) the probability is 0.58×10^{-4} for the eruptive scenario and 1.6×10^{-4} for the groundwater release scenario. Figure 3-3 shows the mean calculated for the 5,000

realizations and the 5th, 50th and 95th percentile dose rates (the 5th percentile curve is barely seen at the lower right-hand edge of the figure).

The mean annual dose in the first few thousand years is dominated by the eruptive release scenario. The mean annual dose from the eruptive release scenario is less than 0.03 mrem/year and diminishes quickly because of radioactive decay of americium-241, the dominant radionuclide for this scenario. Factors taken into account in calculating the mean annual dose for this scenario include:

- Probability of eruption through the repository
- Ash and waste particle sizes
- Wind direction and speed
- Biosphere dose conversion factors (BDCFs) for the eruptive release pathway.

The mean annual dose calculated for the groundwater release scenario increases with time as radionuclides traveling through the rock eventually reach Amargosa Valley. By 3,000 years after closure of the repository, this scenario dominates the overall dose rate. The mean annual dose is less than 0.1 mrem/year in the first 10,000 years and reaches a value of less than 0.2 mrem/year at 50,000 years. The following factors are taken into account in the analysis of this scenario:

- Probability of the intrusive event
- Response of the repository (damage to waste packages and drip shields)
- Unsaturated zone flow and seepage into the emplacement drifts
- Radionuclide concentrations in the water
- Radionuclide delay through the unsaturated zone
- Radionuclide delay through the saturated zone
- Dilution of radionuclide concentrations during pumping
- BDCFs for the groundwater release pathway.

Expected performance of the repository is estimated by adding the mean annual dose for the nominal and igneous activity scenarios. The mean annual dose in the first 10,000 years for the combined nominal and igneous activity scenarios is less than 0.1 mrem/year (CRWMS M&O 2000h, Section 4.3), well below the radiological exposure limit of NRC's proposed regulation.

3.1.3 Total System Performance Assessment for the Inadvertent Human Intrusion Scenario

NRC's proposed regulation defines a specific scenario to be used to evaluate the robustness of the repository system with respect to inadvertent human intrusion (see Appendix B). The proposed regulation specifies an exploratory borehole drilled to the water table 100 years after repository closure. The borehole is specified to penetrate a single waste package (as well as the overlying rock, drip shield and other engineered barriers, and the underlying rock). The analysis of this scenario is presented in Section 4.4 of the report of the current TSPA analyses (CRWMS M&O 2000h). The approach to this analysis is summarized in Appendix C. The results show the mean annual dose is less than 0.01 mrem/year and no realization results in a mean annual dose exceeding 0.5 mrem/year in the first 10,000 years (CRWMS M&O 2000h, Section 4.4.2).

The peak dose rate in this specified scenario is most affected by the following:

- The amount of seepage contacting the waste intersected by the borehole
- Radionuclide concentrations in this seepage
- Delay of radionuclide migration through the saturated zone
- Dilution of the radionuclide concentrations during pumping
- BDCFs for the groundwater release pathway.

These factors are included among the factors considered in the previous sections for the nominal and igneous activity scenarios.

3.2 DEGRADED BARRIER ANALYSES

Analyses have been conducted for selected parameter variations to investigate the sensitivity of the mean annual dose to those parameters. The analyses are conducted for variations in the TSPA component model parameters within their expected range. These analyses therefore do not serve to determine the total contribution of a factor to the estimate of mean annual dose. For example, if the expected range for the factor is small, the analysis would show no significant effect in taking it to the limit of this range, even if the factor were to dominate system performance. The analyses serve to investigate the relative importance of uncertainty in the performance of the barrier. The results of these analyses are reported in Sections 4.6, 5.2, and 5.3 of the report of the current TSPA analyses (CRWMS M&O 2000h). Specifics of the individual analyses can be found in the table in Appendix G of this reference.

The results of degraded barrier analyses are shown in Figures 3-4 through 3-15. These analyses are conducted to explore the importance of uncertainties in the performance of various barriers. The analyses do not explore the full range of uncertainty but only that associated with degraded performance of barriers. The object is to explore whether degradation of a barrier within the range of uncertainty results in significant increase in the mean annual dose estimate. These analyses do not serve to determine the total contribution of the barrier to the estimate of mean annual dose. For example, if the expected range of performance of the barrier is small, the analysis will show no significant effect in taking it to the limit of this range, even if the barrier is the dominant contributor to system performance. The analyses merely serve to investigate the relative importance of uncertainty in the performance of the barrier with respect to meeting the postclosure performance objective.

The base case in these analyses is the mean annual dose for the nominal scenario. The approach in each analysis is to fix selected parameters to specified values while the other parameters are treated as in the base-case analysis. To assess the effect on the mean annual dose of the uncertainty in the specified parameters, the selected parameters are set to their 5th or 95th percentiles, whichever would result in the higher dose rate. The analyses are probabilistic, with Monte Carlo sampling (100 realizations) conducted for those parameters not fixed at the specified values.

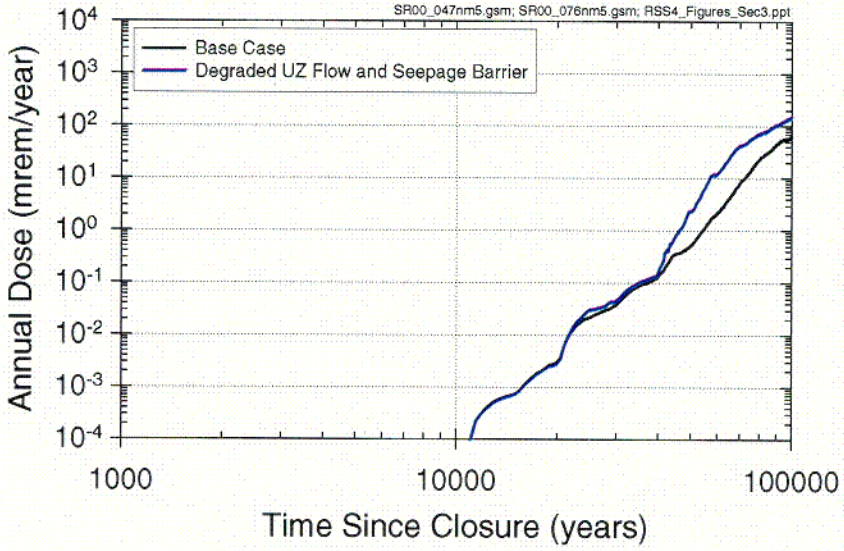

Figure 3-4. Nominal Scenario—Degraded Performance of the Unsaturated Zone (UZ) Flow and Seepage Barrier

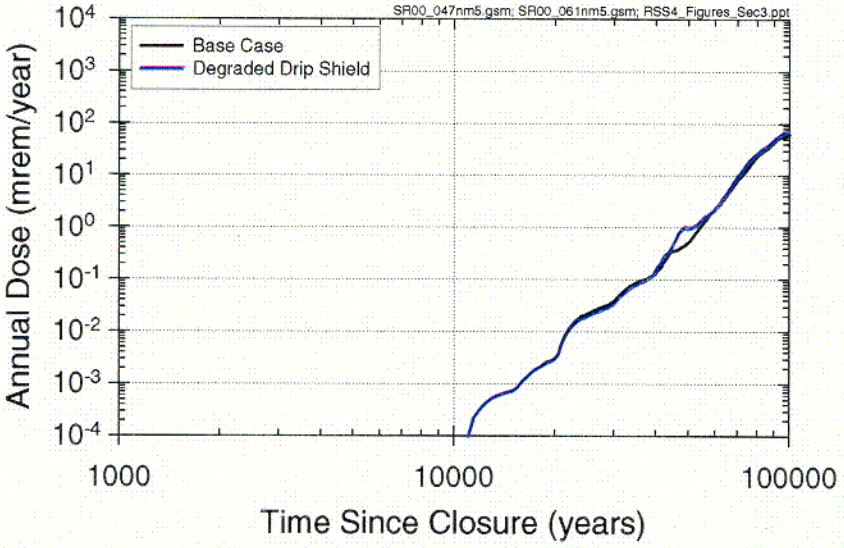

Figure 3-5. Nominal Scenario—Degraded Performance of the Drip Shield

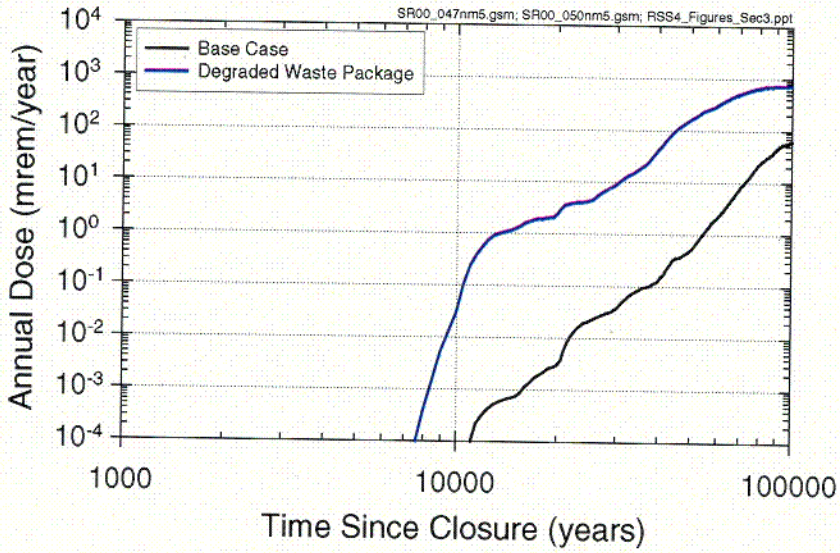

Figure 3-6. Nominal Scenario—Degraded Performance of the Waste Package

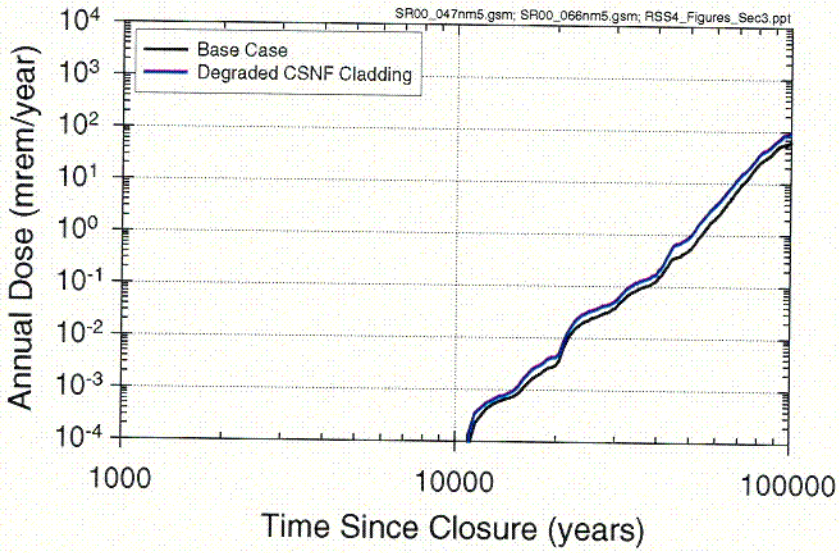

Figure 3-7. Nominal Scenario—Degraded Performance of the CSNF Cladding

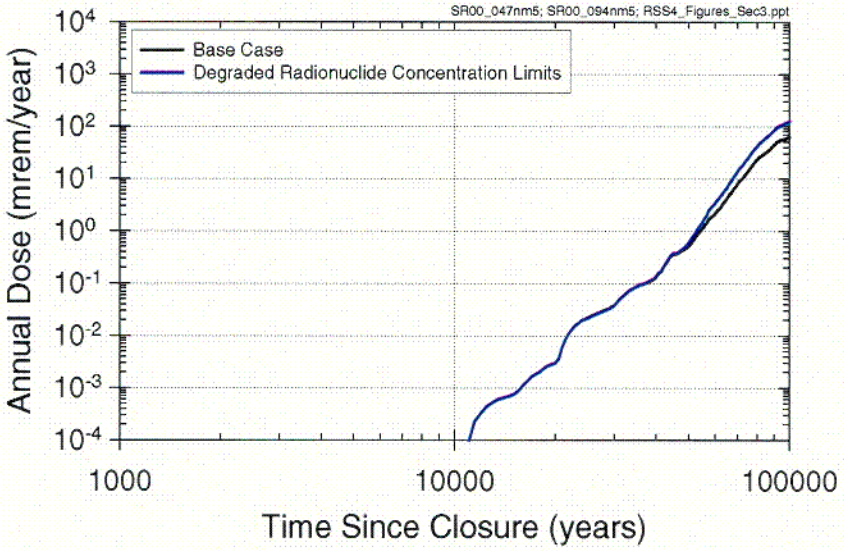

Figure 3-8. Nominal Scenario—Sensitivity to Radionuclide Concentrations Limits in Water

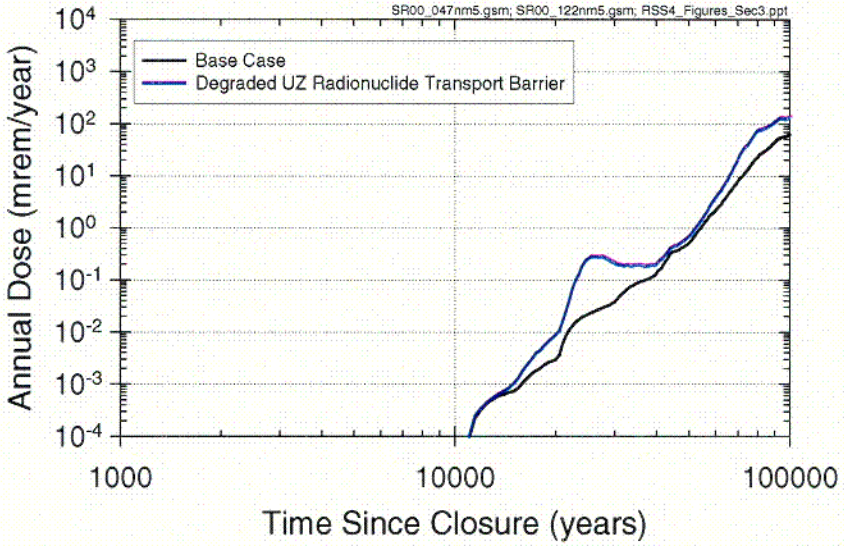

Figure 3-9. Nominal Scenario—Degraded Performance of the Unsaturated Zone (UZ) Radionuclide Transport Barrier

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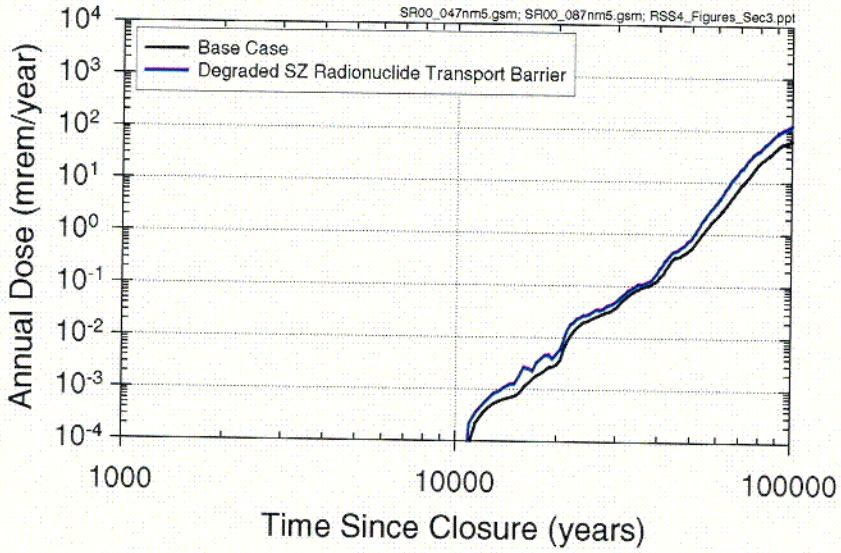

Figure 3-10. Nominal Scenario—Degraded Performance of the Saturated Zone (SZ) Radionuclide Transport Barrier

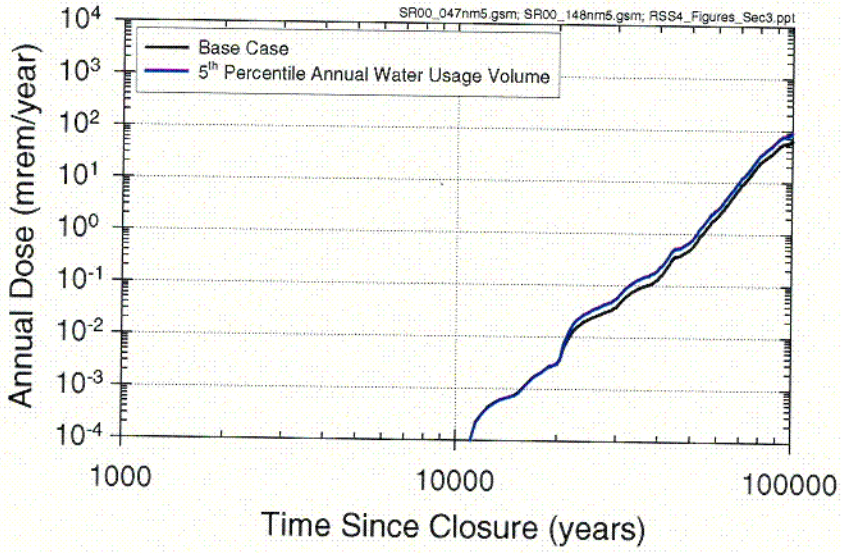

Figure 3-11. Nominal Scenario—Sensitivity to Annual Water Usage Volume

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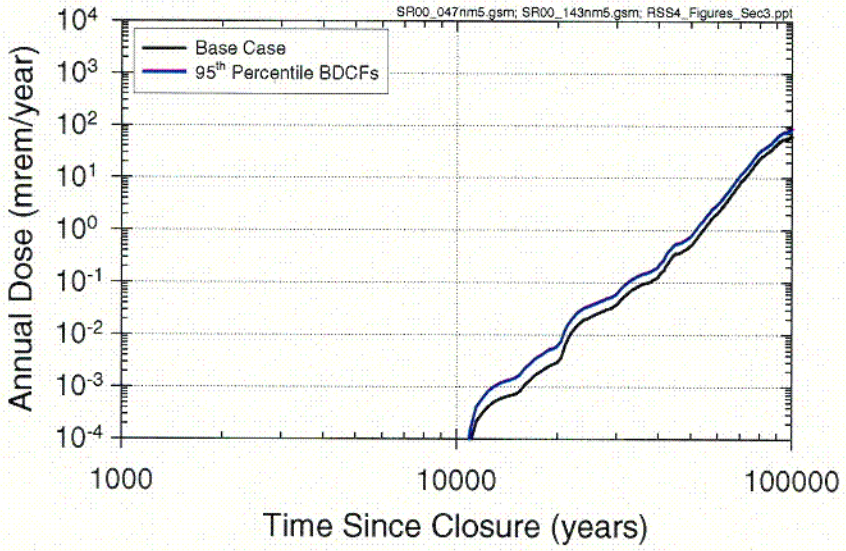

Figure 3-12. Nominal Scenario—Sensitivity to Biosphere Dose Conversion Factors (BDCFs)

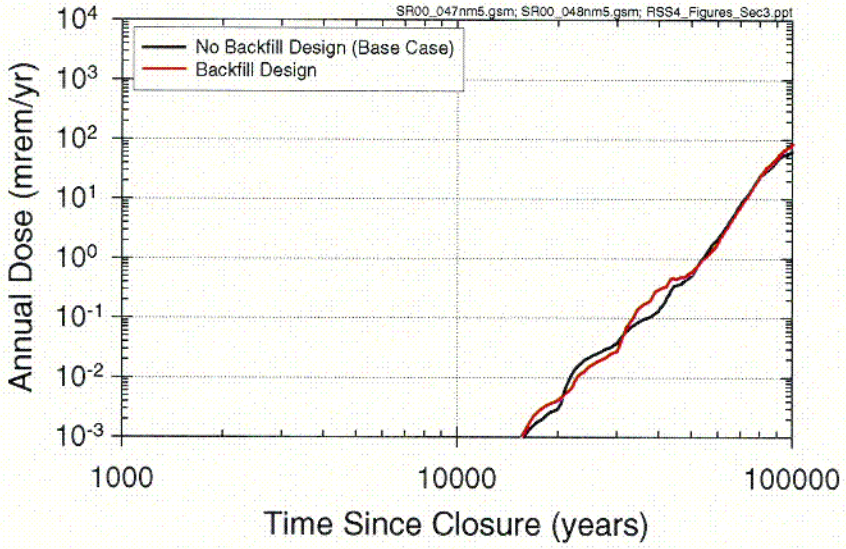

Figure 3-13. Nominal Scenario—Sensitivity to Temperatures in the Emplacement Drifts (Backfill vs. No Backfill Design)

C08

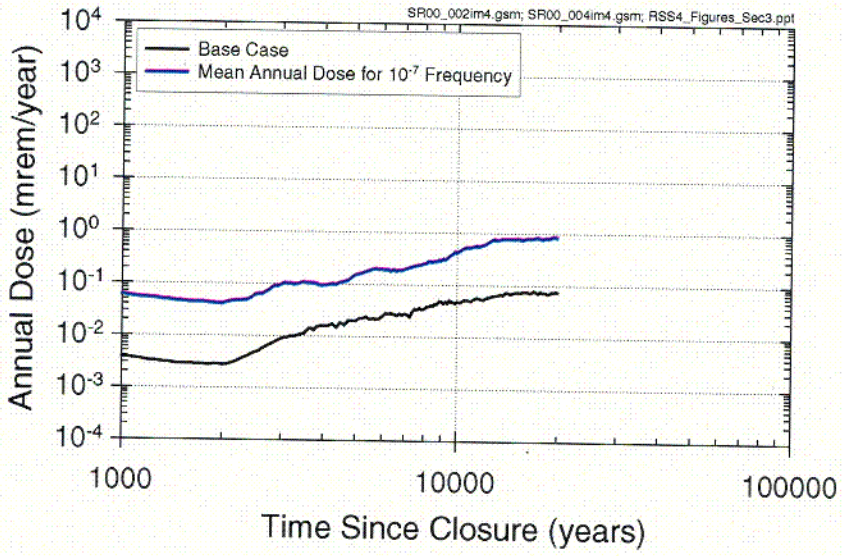

Figure 3-14. Igneous Intrusion Groundwater Release Scenario—Sensitivity to Probability of Igneous Intrusion

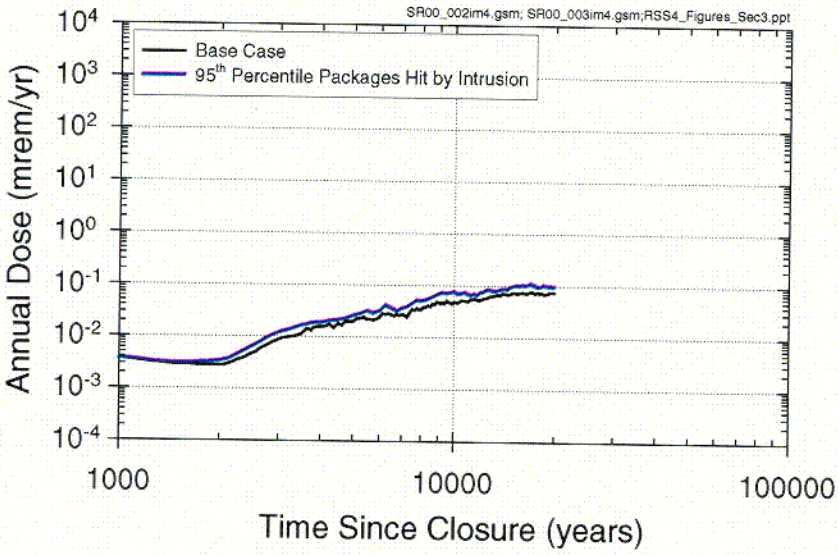

Figure 3-15. Igneous Intrusion Groundwater Release Scenario—Sensitivity to Effects of Igneous Intrusion on Waste Packages and Drip Shields

C09

The difference between the mean annual dose in each calculation and the mean annual dose for the base-case analysis indicates the potential effect of degrading the barrier in the way specified. The specific parameter changes for each case are provided in Appendix G of the report of current TSPA analyses (CRMWS M&O 2000h). The parameters considered in a specific analysis can be seen in the element of the table in this appendix associated with the identification number above the frame for the associated figure shown here.

Figure 3-4 shows the degraded barrier analysis for the unsaturated zone flow and seepage barrier. The analysis is conducted by fixing parameters associated with infiltration, flow, and seepage to the 95th percentile values within the range of known uncertainty. Comparison with the base-case estimate of expected performance shows no significant effect in the first 40,000 years, because the waste packages remain substantially intact during this period, even under the increased flow conditions. Only small cracks develop in the waste packages in this period and release is limited to a small amount of diffusion through them. After about 40,000 years wider breaches develop due to general corrosion. The releases are largely controlled by concentration limits for the radionuclides. However, the increase in seepage in the degraded barrier case results in only modest changes even after significant degradation of the waste package and drip shields has occurred.

Figure 3-5 shows the degraded barrier analysis for the drip shield. The analysis is conducted by setting the drip shield degradation rate to 95th percentile degradation value. The results show little sensitivity to this change, reflecting that performance of the waste package still prevents release of radionuclides as long as the drip shield remains intact.

Figure 3-6 shows the degraded barrier analysis for the waste package. This analysis is conducted by selecting 95th percentile values for the number of initial waste package defects (i.e., defects at the time the waste is emplaced), stress, weld yield strength, and degradation rate. The percentage change in the estimate of expected performance is large, indicating the importance of the waste package uncertainty to overall system performance uncertainty.

Figure 3-7 shows the degraded barrier analysis for CSNF cladding. Performance of the cladding is determined by its initial defects (i.e., defects at the time the waste is emplaced) and degradation of the cladding after emplacement. In the base case a fraction of the cladding in each CSNF waste package is perforated by the time the waste package breaches. In the sensitivity analysis, the cladding is assumed to be further degraded—initial defects of the cladding and cladding degradation rates are taken at their 95th percentile values. Although not all waste (e.g., HLW) includes cladding, results still show some sensitivity—an increase of a factor of less than two.

Figure 3-8 shows the sensitivity of the mean annual dose to parameters governing the concentration limits of radionuclides in water. The calculational approach is to set the chemistry within the waste package and drift invert to aggressive values and to set radionuclide solubility limits and concentrations of radionuclides sorbed irreversibly to colloids to 95th percentile values. The results show negligible changes in the first 50,000 years and minor changes after this time.

Figures 3-9 and 3-10 show the degraded barrier analysis for the unsaturated and saturated zone radionuclide transport barriers. The approach in these analyses is to set radionuclide retardation factors (those derived from sorption of dissolved and colloidal concentrations and from matrix diffusion) in each case to the 5th percentile values in their expected range. Comparison with the base case results for each case indicates modest sensitivity.

Figure 3-11 shows the sensitivity of the mean annual dose to the water usage volume associated with pumping of water in Amargosa Valley. The calculated dose rate is inversely proportional to the dilution volume. Because the variation of the pumping volume in the base-case analysis is small, the variation in mean annual dose is small. The sensitivity to this factor over its range is negligible.

Figure 3-12 shows the sensitivity of the mean annual dose to biosphere transport and uptake (represented by the BDCFs). The proposed regulation dictates that the biosphere model for the TSPA analyses is to be constrained to present-day conditions (eating habits, water intake, etc.). The figure indicates little sensitivity to this factor over the range of variability in present-day conditions.

Figure 3-13 compares the mean annual dose for the nominal scenario discussed in Section 3.1 with the result for a design in which the emplacement drifts are backfilled at the time of permanent closure (the base case of all of these analyses does not include backfill). The main change taken into account is the effect of increased drip shield, waste package, and waste temperatures arising from the insulating effect of the blanket of backfill. In-drift temperatures are increased by the addition of backfill. For example the average peak temperature of the waste package increases from 160°C to 280°C (CRWMS M&O 2000h, Section 4.6). Even so, the results with and without backfill are not significantly different. One reason for this small difference is that waste package performance is essentially the same in the two cases; the degradation rate of the waste package is not strongly affected by temperature within the range spanned by this difference and, consequently, no waste package fails before 10,000 years in either case.

Likewise, other effects of the temperature (e.g., change in water chemistry and radionuclide solubility) are not shown to be significant. To some extent, the lack of sensitivity to in-drift temperatures reflects the conservative nature of the models. For example, reduced ability for radionuclides to diffuse out of the waste package because of removal of water by heat is not taken into account in these models. If this effect were considered, the analysis for the case with higher temperatures could be different and could show a different degree of sensitivity.

Sensitivity analyses of aspects of the igneous activity scenario are not shown directly here, but are shown in Section 5.2.9 of the report of current TSPA analyses (CRWMS M&O 2000h). However, because the most important contributor to the mean annual dose for this scenario is by way of the groundwater release pathway, conclusions about the sensitivities would be the same as those drawn for the nominal scenario, which also involves the groundwater release pathway. Consequently, conclusions drawn from Figures 3-4 through 3-12 are considered to apply to the igneous activity scenario.

The sensitivity studies for the nominal scenario do not explore the sensitivity to the probability of the igneous activity scenarios. Consequently, the sensitivity to the probability of occurrence for the igneous activity scenarios cannot be inferred from any of those sensitivity studies. The sensitivity to this factor is shown in Figure 3-14. In this case the mean annual dose calculated for the igneous activity scenarios is compared with the case calculated for a mean annual frequency of 10^{-7} per year (applied to both the eruptive scenario and the groundwater release scenario). The comparison shows an increase in the estimated mean annual dose essentially proportional to the increase in frequency of about a factor of six.

Sensitivity to the number of waste packages and drip shields that are damaged by magmatic intrusion is evaluated in Figure 3-15. The effect is small because the variation in the number is small.

3.3 NON-MECHANISTIC JUVENILE FAILURE SCENARIO ANALYSES

The degraded barrier analyses in the previous section show limited changes before 20,000 years, because waste packages remain intact and prevent release of radionuclides. To investigate these other component models more completely, studies have been conducted with the waste exposed at early times. The scenario evaluated is a non-mechanistic juvenile failure scenario in which a waste package is artificially set to fail at 100 years after repository closure. This failure is assumed only to gain insight into the possible sensitivity of factors obscured by the performance of the waste packages. Results are shown in Figure 3-16 for the cases of non-mechanistic juvenile failure of a single CSNF waste package and a single co-disposal waste package containing HLW.

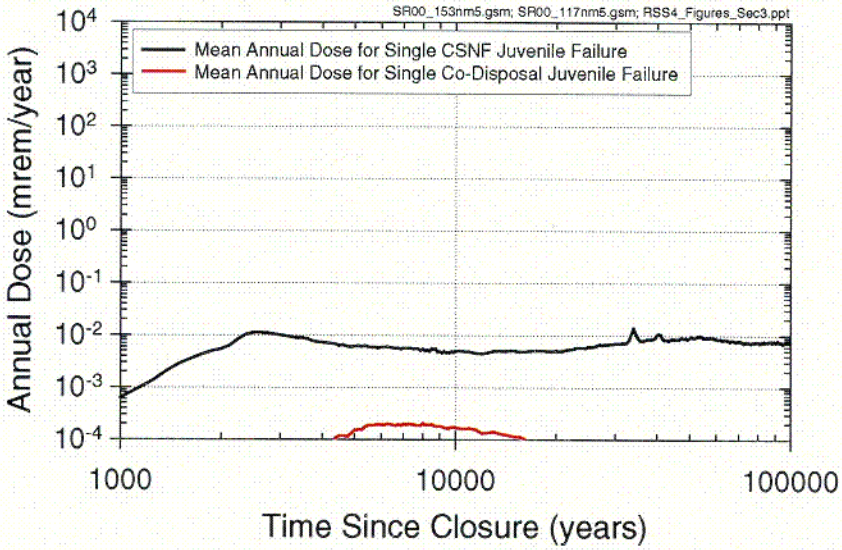

Figure 3-16. Non-Mechanistic Juvenile Waste Package Failure Scenario Base Case Results

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The analysis in each case considers only the single waste package. To ensure a large juvenile breach, the analysis imposes a large number of pits through the waste package at 100 years after emplacement, approximating them with a patch breach of 300 cm^2 through the waste package wall. The drip shield remains intact until it fails because of expected degradation processes. The details of the parameters used for this analysis are reported in Appendix G of the report of current TSPA analyses (CRWMS M&O 2000h).

For both cases in Figure 3-16, initial release from the system occurs in less than 10,000 years. The peak dose rate in each case is small, approximately 0.01 mrem/year, and arises almost entirely from technetium-99, iodine-129, and neptunium-237. Comparison indicates that the release from the CSNF waste package is higher than that from the co-disposal waste package by approximately an order of magnitude. This result arises from the differences in radionuclide inventories in the two different packages and from the differences in temperature and chemistry within the two waste packages.

The analysis for the CSNF waste package provides the base case for barrier degradation analyses for this non-mechanistic juvenile waste-package failure scenario. These analyses explore the same cases as considered previously for the nominal scenario (in which the waste package performs as expected). The approach to the specific parameter variations for this juvenile failure scenario is the same for those analyses, the only difference being the large breach in the waste package in this scenario. The analyses set values of selected parameters to their 95th or 5th percentile values while maintaining the other parameters for this scenario at their base-case values or probability distributions. The specific parameter changes for each case are provided in Appendix G of the report of current TSPA analyses (CRMWS M&O 2000h). The results of the sensitivity studies for the CSNF juvenile waste-package failure scenario are shown in Figures 3-17 through 3-22.

Figure 3-17 shows little sensitivity of the juvenile failure waste-package performance to seepage into the emplacement drifts in the base case model for 100,000 years. In this period, neptunium-237 is the largest contributor to the mean annual dose in both the base case and this sensitivity analysis. Advective release of this radionuclide in the model is controlled by the amount of water entering the waste package through the assumed breach and the concentration of the radionuclides in this water. Other than the dependence of the drift invert properties on moisture content, the model for diffusive release is controlled only by the concentration of the radionuclide and not the amount of water. Diffusion is determined by the concentration gradient, which is, in turn, determined by the concentration of the radionuclides, and not the amount of water. Consequently, that the results are not sensitive to the amount of seepage indicates the release from the engineered barriers is dominated by diffusion.

The base model for diffusion away from the waste package and through the drift invert is conservative (CRWMS M&O 2000h, Section 3.6.3.1). For example, the model does not treat the invert as composed of granular material but as a homogeneous medium with a uniform diffusion coefficient that is a function of the moisture content of the invert as a whole. Further, the base-case diffusion coefficient is conservative. Measurements by Conca and Wright (1992, p. 22) suggest the diffusion coefficient may be less than 10^{-11} cm^2/sec when flow through the invert is negligible. These results need to be evaluated; to investigate the impact of the diffusion

coefficient, the sensitivity analyses were repeated with the diffusion coefficient set to this low value.

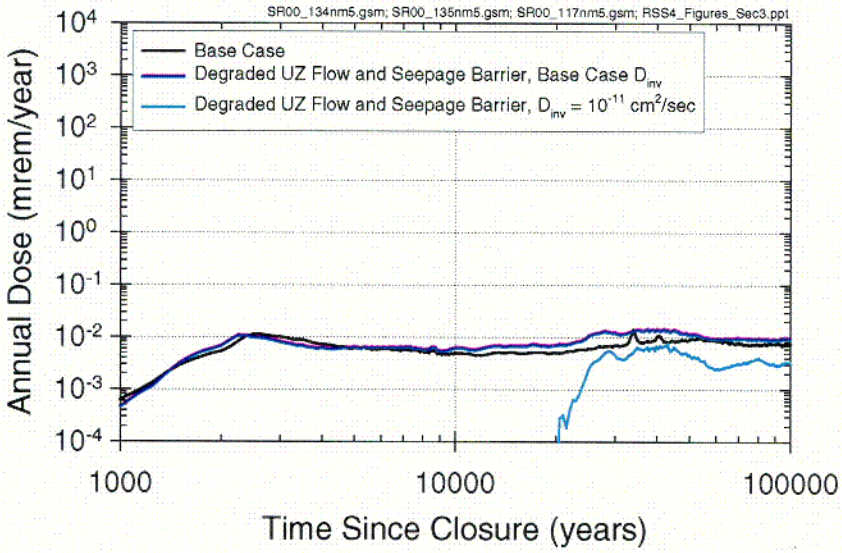

Figure 3-17. Non-Mechanistic Juvenile Failure Scenario—Degraded Performance of the Unsaturated Zone (UZ) Flow and Seepage Barrier

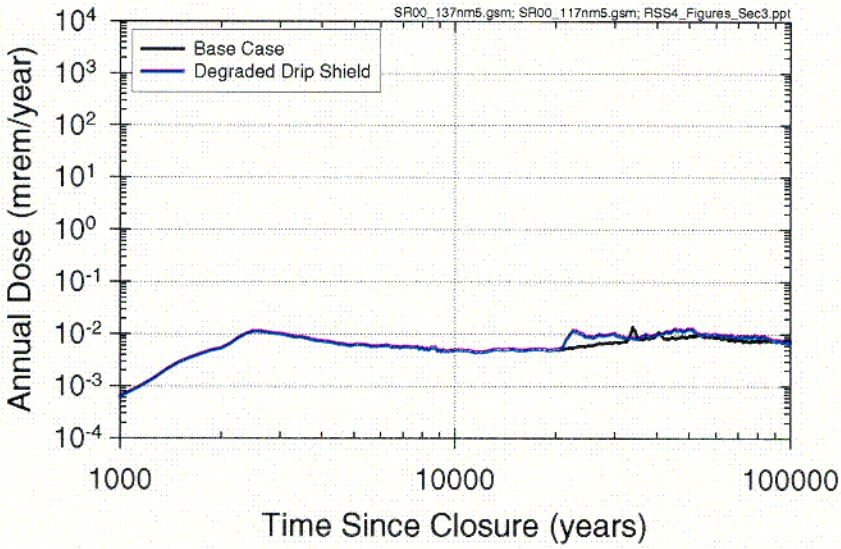

Figure 3-18. Non-Mechanistic Juvenile Failure Scenario—Degraded Performance of the Drip Shield

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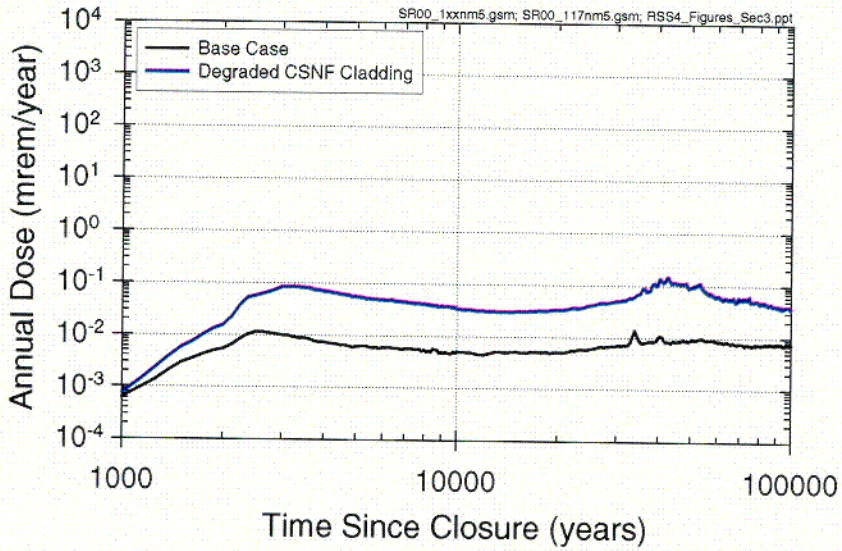

Figure 3-19. Non-Mechanistic Juvenile Failure Scenario—Degraded Performance of CSNF Cladding

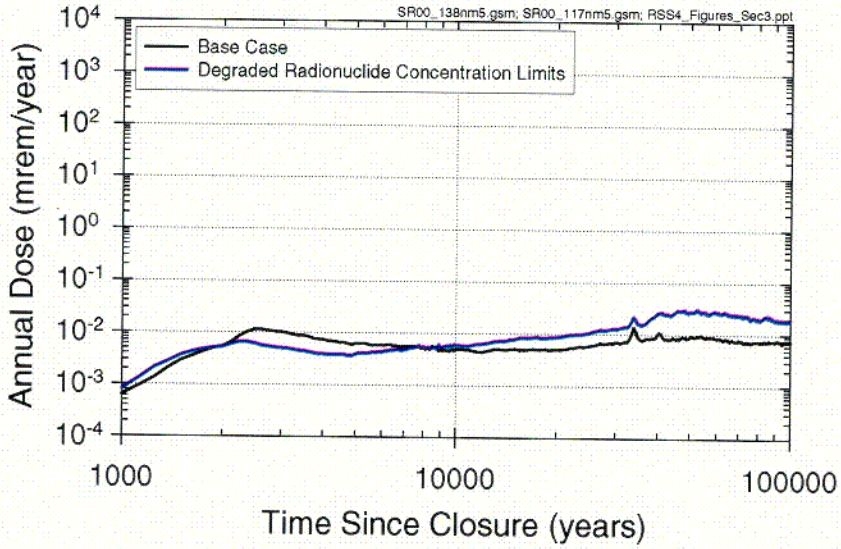

Figure 3-20. Non-Mechanistic Juvenile Failure Scenario—Sensitivity to Radionuclide Concentration Limits

C12

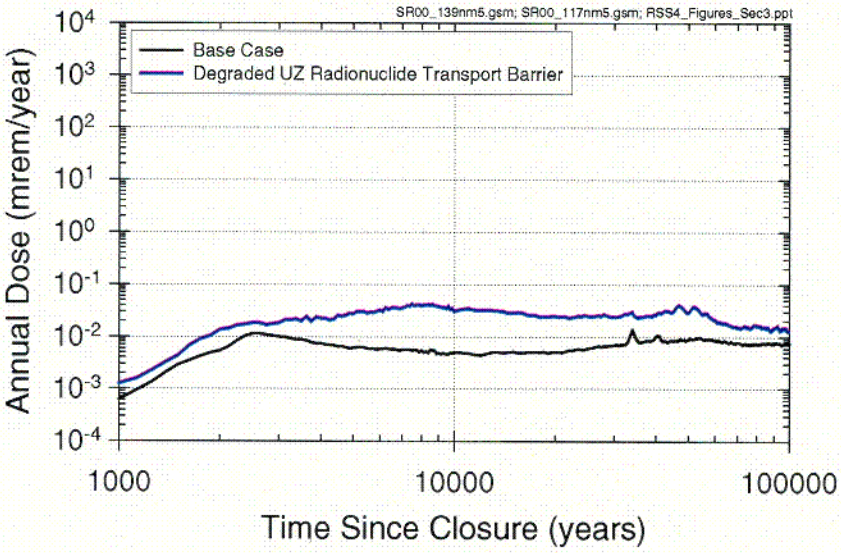

Figure 3-21. Non-Mechanistic Juvenile Failure Scenario—Degraded Performance of the Unsaturated Zone (UZ) Radionuclide Transport Barrier

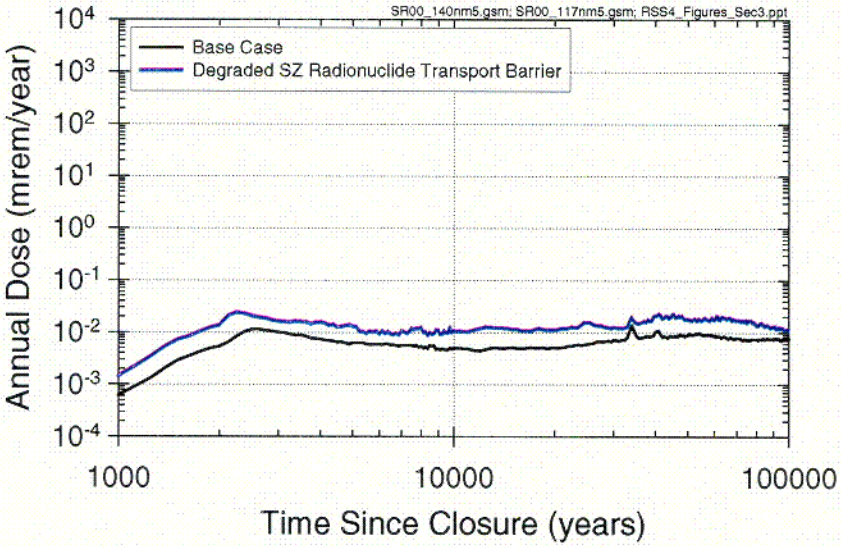

Figure 3-22. Non-Mechanistic Juvenile Failure Scenario—Degraded Performance of the Saturated Zone (SZ) Radionuclide Transport Barrier

C13

The results for the calculation using the lower diffusion coefficient indicate the mean annual dose is significantly reduced in the first 20,000 years while the drip shield remains intact. Simply reducing the diffusion coefficient dramatically lowers the release, which indicates that diffusive release is dominating the results during the first 20,000 years. The effect of lowering the diffusion coefficient decreases with time, indicating the increasing importance of advective release as the number of failed drip shields increases. However, a small difference exists even at 100,000 years, indicating that diffusive release tends to be more important than the advective component, even at this time. This result reflects the conservative nature of the diffusive transport model in the base case and confirms the conclusion drawn from the lack of sensitivity to the degree of seepage in the emplacement drift. These results indicate the potential importance of the diffusion model to expected performance.

Figure 3-18 shows the sensitivity to performance of the drip shield. The lack of sensitivity to the drip shield performance in this calculation results from the fact that the release of radionuclides from the engineered barrier system is dominated by diffusion rather than advection. The drip shield serves only to preclude flow and therefore only affects the advective transport component. The lack of sensitivity to the drip shield lifetime therefore reflects the minor role of the advective component of transport through the drift invert. If the diffusion coefficient of the drift invert were significantly reduced, sensitivity to performance of the drip shield would be greater.

These results indicate the important role the drift invert plays in determining the importance of the drip shield. These results indicate the benefit of considering the performance of these two elements together rather than as separate elements. In Section 4, where the principal factors are defined, performance of the drip shield and drift invert is combined into a single factor.

Figure 3-19 shows the sensitivity to performance of the CSNF cladding in the event of juvenile failure of the CSNF waste package for the first 100,000 years. The base case assumes that a small fraction of the cladding is breached at emplacement and that the remainder of the cladding degrades slowly over time according to expected rates. In the sensitivity analysis, a much larger fraction of the cladding is breached at emplacement and the degradation rate is much higher. The increase in the mean annual dose resulting from increased initial cladding breaches and increased degradation rate is approximately a factor of ten. This result can be compared to the result for the analogous sensitivity study for the nominal scenario (see Figure 3-8). In that case, the same variation in CSNF cladding performance produced a factor of less than two increase in the mean annual dose. The difference between the results of these two sensitivity analyses arises in part because the first analysis includes the effects of all waste packages, including those containing HLW. The latter waste does not include cladding; consequently changes in cladding properties have no effect on them. Consequently, these changes produce a smaller effect on total system performance when all the waste packages are taken into account.

Figure 3-20 shows the sensitivity to the radionuclide concentrations in water for the first 100,000 years. As in the nominal scenario, the change in radionuclide concentration is accomplished by modifying the water chemistry and the radionuclide solubility limits and concentrations of radionuclides irreversibly sorbed to colloids to values that result in increases in those concentrations. However, as in the nominal scenario sensitivity analysis, the mean annual dose is not significantly affected.

Figures 3-21 and 3-22 show the sensitivity to the delay properties of the unsaturated zone and saturated zones. As in the sensitivity study for the nominal case, the effect in each case is small.

3.4 NEUTRALIZED BARRIER ANALYSES

The role of a factor (e.g., the contribution of a barrier) in the estimate of expected performance requires comparison of that estimate with one in which the factor is omitted from the analysis. Analyses in which a particular factor is omitted are referred to as "neutralization" analyses (see Appendix C). These are specialized sensitivity studies in which selected parameters are set to values corresponding to omission (i.e., neutralization) of the factor. The representations of the other factors are the same as in the base-case analysis. The approach to neutralization analyses is summarized in Appendix C.

The neutralization analysis approach applies to barrier functions (e.g., the role of the barrier in limiting radionuclide migration) and to more general factors (e.g., radionuclide concentration limits). The analyses are used to explore the contribution of a factor, not to evaluate performance under low probability conditions. In this case, the analyses are used to gain insight into the TSPA analyses. These analyses are also used to evaluate the contribution of various factors to defense-in-depth. A large change resulting from neutralization of a barrier function would indicate significant reliance on that function and potential limits to defense-in-depth for that barrier.

Neutralization analyses have been conducted for barrier functions and other key factors. Neutralization analyses for the natural barriers, the engineered barriers, and other factors are summarized in the following sections. Details of the model parameters for these neutralization analyses are reported in Appendix G of the report of current TSPA analyses (CRWMS M&O 2000h). Specific parameter changes associated with the neutralizations are given in the element of the table of this appendix associated with the identification number above the frame of each figure presented here.

3.4.1 Neutralization Analyses for Natural Barriers

Neutralization analyses for the natural barriers of the repository system have been conducted to evaluate the function of these barriers in limiting the movement of water or radionuclides. These analyses explore the contribution of the following:

- Unsaturated zone flow and seepage barrier
- Unsaturated zone radionuclide transport barrier
- Saturated zone radionuclide transport barrier.

Figure 3-23 compares the base-case mean annual dose with that calculated with the unsaturated zone flow and seepage barrier neutralized. In this case, the seepage flux into the emplacement drifts is set to the precipitation flux on the surface of the mountain. Other factors such as sorption properties and temperature in the repository are not changed. The results in Figure 3-23 show no significant change to the mean annual dose in the first 40,000 years. To an extent, this reflects that the waste packages are still largely intact; however, the mean annual dose even at 100,000 years is only slightly affected.

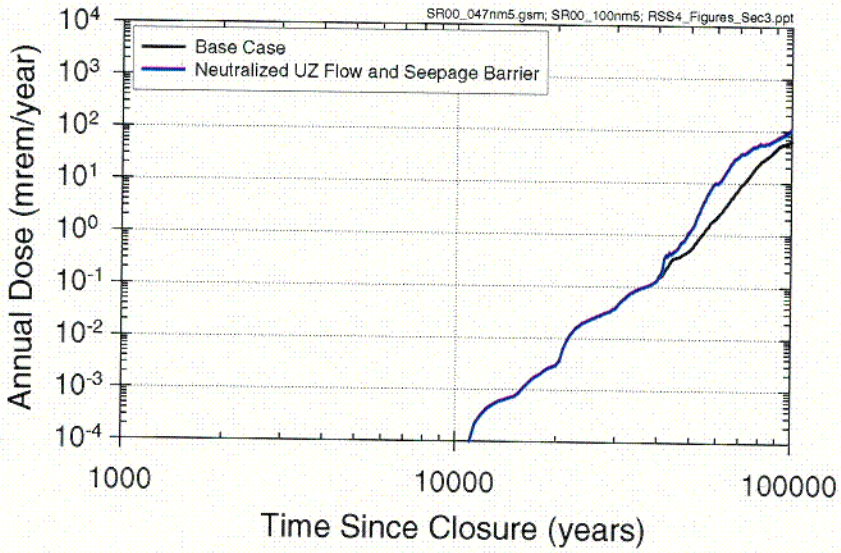

Figure 3-23. Neutralized Performance of the Unsaturated Zone (UZ) Flow and Seepage Barrier

This small change reflects the dominance of diffusive transport in the base-case drift invert transport model. Therefore, analyses using this model show limited sensitivity to the amount of seepage because the radionuclide concentrations, and not the amount of seepage, control the diffusive release.

Figure 3-24 compares the base-case mean annual dose with that calculated with the unsaturated zone and saturated zone radionuclide transport barriers neutralized. This figure shows the effect of neutralizing the unsaturated zone and saturated zone transport barriers separately and the effect of neutralizing both of them at the same time. The approach to each individual neutralization is to short-circuit each barrier. For example, neutralization of the unsaturated zone transport barrier is accomplished by feeding the releases from the engineered barrier system directly into the saturated zone. The neutralization of the saturated zone radionuclide transport barrier is accomplished by feeding the releases from unsaturated zone directly to the biosphere. This approach effectively minimizes the travel time through the respective barriers.

In each case in Figure 3-24, the annual dose in the first 10,000 years is not affected by the neutralization because the waste packages remain intact and prevent release to the natural barriers. However, the dose rate at longer times is affected to some degree in each case. Radionuclides that are substantially delayed in the base case are released sooner when either radionuclide transport barrier is neutralized. The increases shown at about 20,000-30,000 years in the neutralization analysis for the unsaturated zone barrier arises from radionuclides that are delayed much longer in the expected performance case (e.g., americium-243).

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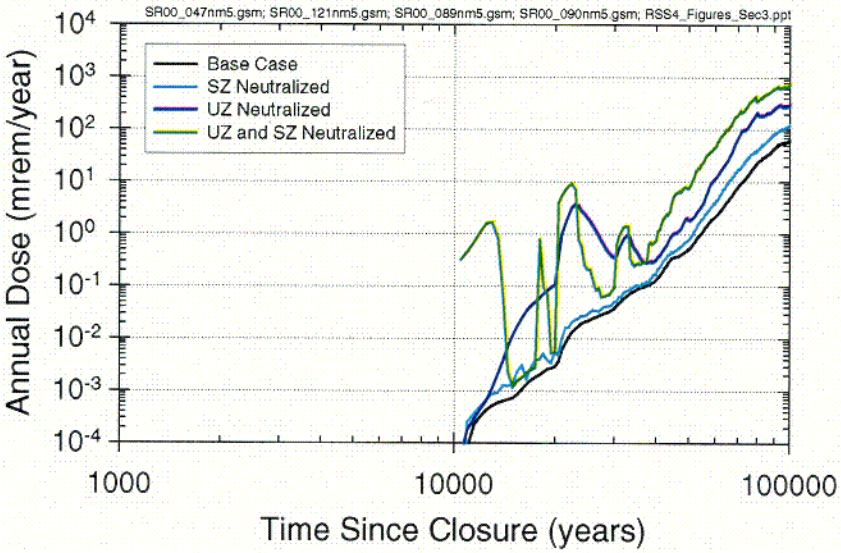

Figure 3-24. Neutralized Performance of Unsaturated Zone (UZ) and Saturated Zone (SZ) Radionuclide Transport Barriers

Figure 3-24 also shows the effect when both of these natural barriers are neutralized. The oscillations in this curve are the result of the averaging over different realizations. The radionuclides that dominate are the same in each case; however, they arrive at Amargosa Valley at different times in the different realizations and the averaging among these different realizations gives this strange looking curve. A larger number of realizations would result in a smoother curve.

3.4.2 Neutralization Analyses for Engineered Barriers

To explore the contributions of the engineered barriers to postclosure performance more fully, analyses similar to those for the natural barriers were conducted for the engineered barriers. The results of these analyses are shown in Figures 3-25 through 3-28. These analyses explore the contribution of the following:

- Drip shield flow barrier
- Waste package flow barrier
- Drift invert radionuclide transport barrier

Figure 3-25 compares the base-case mean annual dose with the mean annual dose calculated with the drip shield neutralized. The approach to this neutralization is to assume that all seepage flux entering the emplacement drifts contacts the waste packages: the drip shield does not divert any water. The net effect of this neutralization is small: the waste packages remain intact and continue to prevent exposure of the waste to water.

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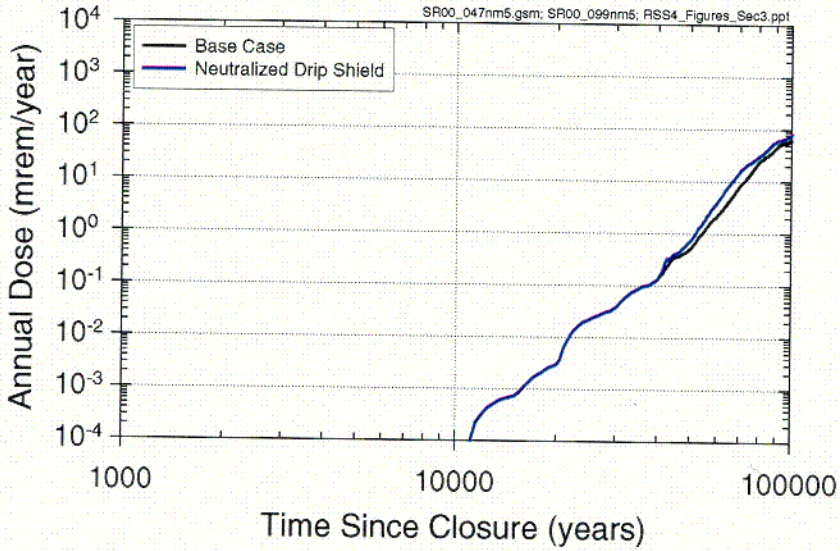

Figure 3-25. Neutralized Performance of Drip Shield

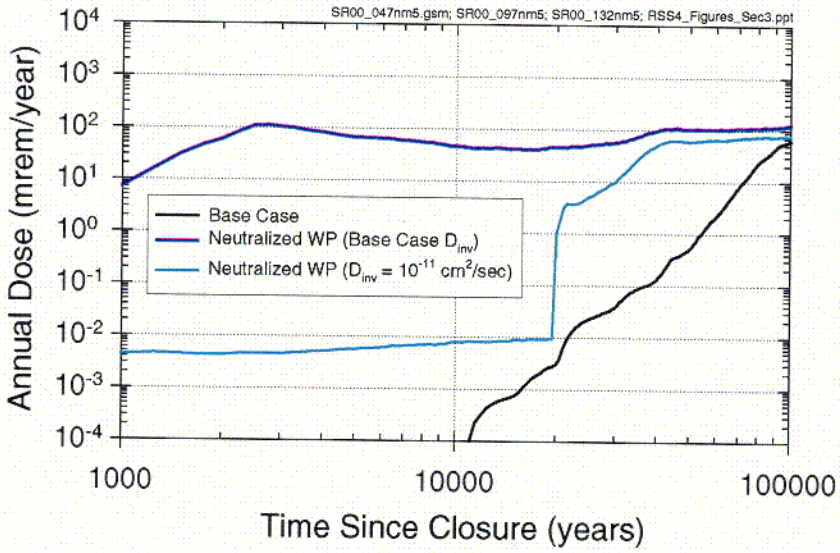

Figure 3-26. Neutralized Performance of Waste Package (WP)

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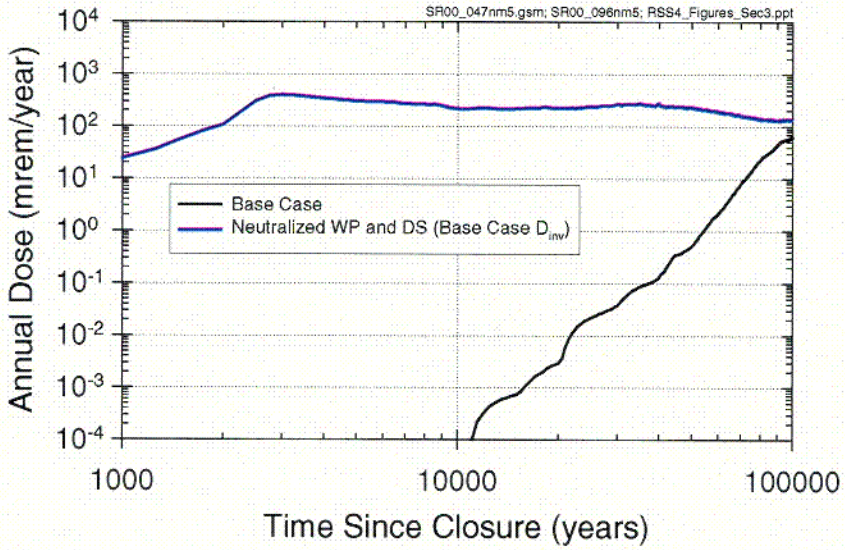

Figure 3-27. Neutralized Performance of Drip Shield (DS) and Waste Package (WP) Together

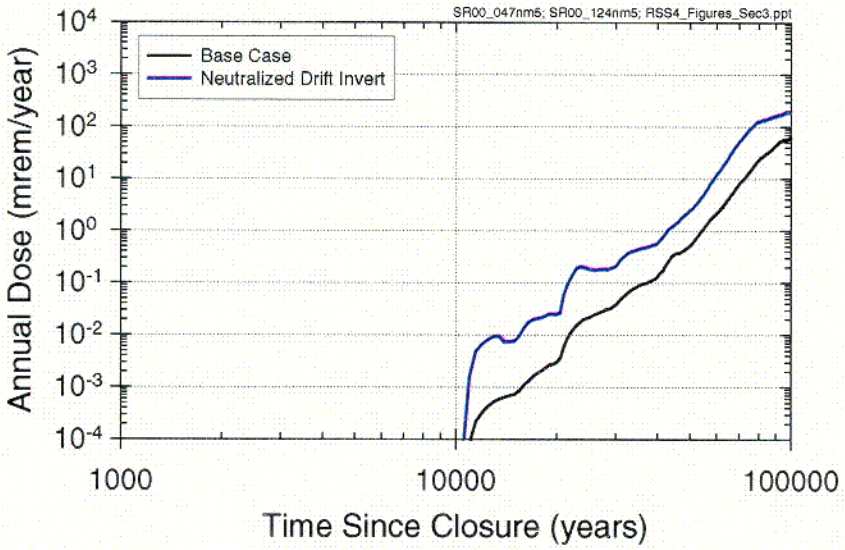

Figure 3-28. Neutralized Performance of Drift Invert Radionuclide Transport Barrier

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Figure 3-26 shows the base case and the mean annual dose calculated with all waste packages neutralized. The calculational approach in this case is the same as for the non-mechanistic juvenile failure scenario. That is, in order to simulate a large number of pits penetrating the waste package wall, a 300 cm^2 breach is imposed at 100 years after permanent closure. The difference in this case, however, is that all waste packages are so breached, and not just a single waste package. A larger breach could have been assumed; however, a failure of this magnitude in every waste package is assumed to be sufficient for the purposes of these analyses.

The change in the mean annual dose for the waste package neutralization is the largest of any of the single-barrier neutralization analyses. This result reinforces the importance of the waste package performance as indicated in the base case analysis. The peak dose rate calculated for this neutralization analysis is on the order of 100 mrem/year. This result, calculated for early failure of every waste package, indicates the important role of the other barriers of the system. For example, compared with the result of the case in which no barrier is effective (see the upper curve in Figure 3-2), this case shows the other barriers reduce the mean annual dose by more than seven orders of magnitude.

Although, the mean annual dose in this neutralization does exceed the radiological exposure limit of the proposed rule, this result should not in any way indicate the repository system would fail to meet the performance objective of the proposed rule. That is, this neutralization does not represent a credible situation because the probability that all packages would breach in the first 100 years (i.e., be neutralized) is negligible (see Sections 3.2.7.1 and 3.2.7.2 of CRWMS M&O 2000c). The purpose of the analysis is to gain insight into the contribution of various barriers to the mean annual dose. The insight in this case is that the barriers of the system without the benefit of the waste package contribute substantially to repository performance—these barriers alone limit the mean annual dose to less than 100 mrem/year, below the level associated with natural background radiation (NCRP 1987, p. 52).

Even this calculated mean annual dose might be greater than would actually occur. As discussed for the juvenile failure scenario, the primary mode of radionuclide release from breached waste packages is diffusive transport through the drift invert. That diffusive release dominates is expected while the drip shield remains intact and prevents advective flow. However, in the conservative model for the diffusive transport in the base case, this is true even after the drip shield is breached in these calculations. This point is explored by repeating the neutralization analysis with the lower diffusion coefficient considered in the juvenile failure scenario analysis. The result is that, while the drip shield remains intact, the release is very small because not only is advective release eliminated in this period, diffusive release is very small. After the drip shield breaches, the mean annual dose increases because advective release is then permitted. This analysis reinforces the potential benefit of the drip shield and drift invert working together to provide defense-in-depth in the event of early failure of a waste package.

Figure 3-27 compares the base-case mean annual dose with that calculated with the waste package and drip shield barriers neutralized simultaneously. The invert diffusion model for the neutralization analysis corresponds to the base case model. The effect of the neutralization is significant, more so than for either of the neutralizations of the waste package or drip shield separately. This analysis indicates that the drip shield backs up the waste package and provides defense-in-depth by limiting release of radionuclides from the engineered barrier system.

Figure 3-28 compares the base case mean annual dose with that calculated with the drift invert transport barrier neutralized altogether (but the waste package and drip shield remaining intact). The calculational approach in this case is to set the saturation in the invert to 100 percent. The net effect of the neutralization is small in comparison with the radiological exposure limit of the proposed regulation. As already discussed, the model for the drift invert transport properties is very conservative; consequently, changes in these properties do not produce a large change here.

3.4.3 Other Neutralization Analyses

Two other neutralization analyses are considered here. The results of these analyses are shown in Figures 3-29 and 3-30. The first considers neutralization of the CSNF cladding. Figure 3-29 shows the effect of neutralizing the CSNF cladding. The neutralization does not result in any effect in the first 10,000 years because the waste packages remain intact for this period. After the CSNF waste packages begin to fail there is an effect, but the effect is small. The change is larger than for the case in which the cladding is merely degraded (see Figure 3-8), a factor of ten rather than a factor of two.

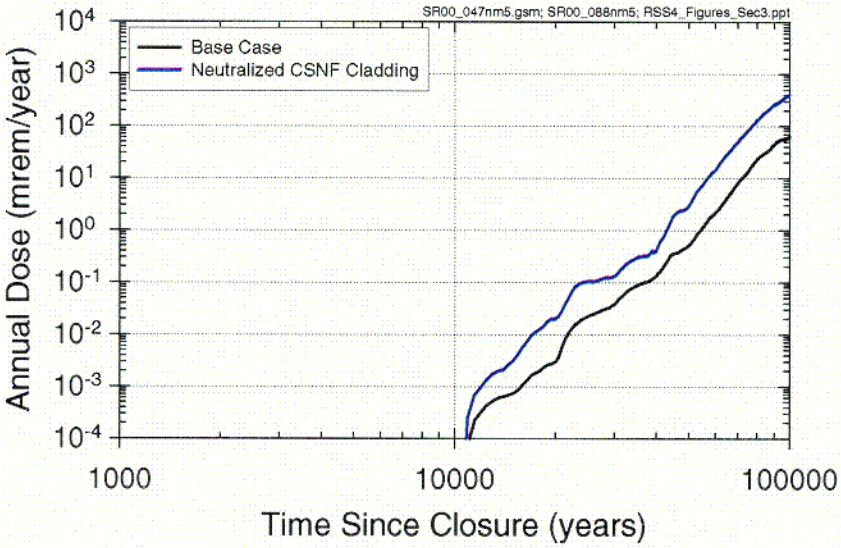

Figure 3-29. Neutralized Performance of CSNF Cladding

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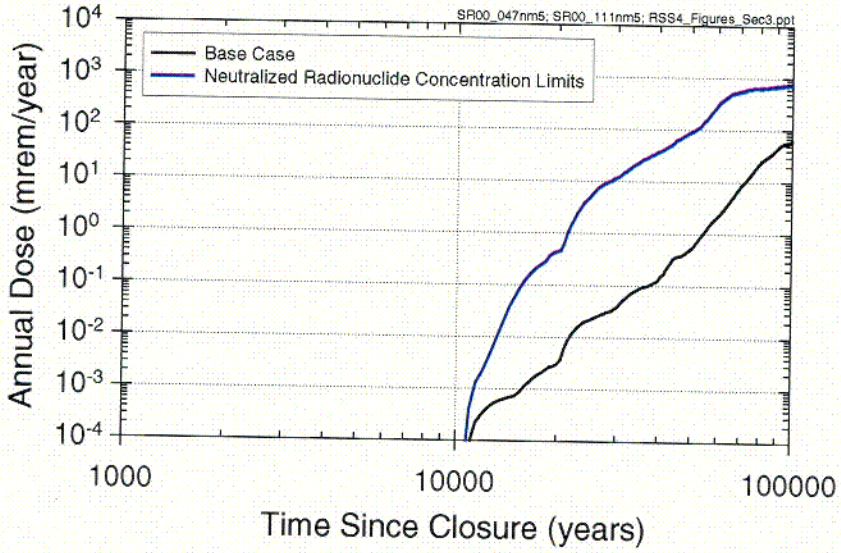

Figure 3-30. Neutralization Analysis of Radionuclide Concentration Limits

Figure 3-30 shows the effect of neutralizing the radionuclide concentration limits in the water. The degraded barrier analyses in Section 3.2 considered the effect of changes in the concentration limits over the range considered to be credible. That effect was shown to be small (see Figure 3-8). In this case, a more extreme change is considered. The solubility limits of all radionuclides are set so high that the only factor controlling the concentrations is the degradation rate of the waste form. The effect is much larger and suggests the concentration limits play a key role in determining the dose rate in the event of breached waste packages.

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4. PRINCIPAL FACTORS OF THE POSTCLOSURE SAFETY CASE

The principal factors are the factors that determine whether the postclosure performance objective is met. Although a wide range of factors must be considered to evaluate suitability of the site for repository development, only the principal factors are essential to the demonstration of postclosure safety. These principal factors enhance the transparency of the postclosure safety case and identify the areas where uncertainty is most important. They therefore provide an underlying framework for the current postclosure safety case summarized in the next section.

Section 4.1 identifies the principal factors and gives the rationale for their identification. Section 4.2 discusses other factors that have been considered in the site recommendation considerations and gives the rationale for them not being identified as principal factors. Section 4.3 summarizes the expected role of the principal factors in waste isolation.

4.1 IDENTIFYING THE PRINCIPAL FACTORS

4.1.1 Considerations in Identifying Principal Factors

The basis for identification of the principal factors is performance assessment. This identification is made in terms of one or more of the following considerations:

1. They contribute substantially to expected postclosure performance.
2. They are a source of uncertainty sufficient to bring into question whether the postclosure performance objective is met.
3. They provide significant defense-in-depth.

The first consideration is addressed by evaluating the contribution of the various factors in neutralization analyses. If the neutralization of a factor changes the mean annual dose in the first 10,000 years by a substantial amount, the factor is identified as a principal factor. For this revision, a change in the mean annual dose in the first 10,000 years greater than ten percent of the radiological exposure limit of the proposed regulation is considered to be a substantial change.

The second consideration addresses the importance of uncertainty. If uncertainty in a particular factor could lead to an estimate of the annual dose that exceeds the performance standard of the proposed rule, that factor is identified in this revision as a principal factor. In particular, if the 95th percentile of the range of uncertainty in the TSPA sensitivity analysis for the factor exceeds this standard, that factor is identified as a principal factor.

The third consideration addresses the contribution of a factor to defense-in-depth. For the purposes of this strategy, neutralization analyses are used to determine which factors provide defense-in-depth. These analyses help determine whether the performance of the system is overly dependent on any single factor.

The principal factors identified from these considerations are listed in Table 4-1. These principal factors are a refinement of those identified in Revision 3 of this strategy and an extension to

address disruptive events. They determine postclosure performance of the repository under scenarios credible at Yucca Mountain. This table shows their relationship to key waste isolation attributes described in Section 2.

Table 4-1. Principal Factors of the Postclosure Safety Case

Waste Isolation Attributes	Principal Factors
Limited water entering emplacement drifts	Seepage into emplacement drifts
Long-lived waste package and drip shield	Performance of the drip shield/drift invert system
	Performance of the waste package
Limited release of radionuclides from the engineered barriers	Radionuclide concentration limits in water
Delay and dilution of radionuclide concentrations provided by the natural barriers	Radionuclide delay through the unsaturated zone
	Radionuclide delay through the saturated zone
Low mean annual dose even considering potentially disruptive events	Probability of igneous activity
	Repository response to igneous intrusion (damage to waste packages and drip shields)
	Additional factors that also apply to the nominal scenario • Seepage into emplacement drifts • Radionuclide concentration limits in water • Radionuclide delay through the unsaturated zone • Radionuclide delay through the saturated zone

The considerations for each of the principal factors are summarized in the following sections. These are addressed within the context of the nominal scenario, the igneous activity groundwater release scenario, and other scenarios.

4.1.2 Nominal Scenario

The principal factors that apply to the nominal scenario (the scenario for nominally expected conditions) are listed in Table 4-2. The first principal factor in this table is seepage into the emplacement drifts. The amount of water contacting the wastes at locations within the repository depends upon the seepage into the emplacement drifts at those locations. Coupled with concentrations of mobilized radionuclides, this factor determines the amount of radionuclides that could be released from breached waste packages. The technical basis for the representation of this factor is described in the Unsaturated Zone Flow and Transport Process Model Report (PMR) (CRWMS M&O 2000b). This reference describes the precipitation and infiltration (CRWMS M&O 2000b, Section 3.5) and the unsaturated zone flow system (CRWMS M&O 2000b, Section 3.7) that define the conditions controlling the amount of seepage. It also describes spatial and temporal distribution of seepage into the emplacement drifts (CRWMS M&O 2000b, Section 3.9).

Table 4-2. Principal Factors—Nominal and Igneous Activity Scenarios

Principal Factors	Nominal Scenario	Igneous Activity Scenario
Seepage into emplacement drifts	Defense-in-Depth	Contribution to Performance
Performance of the drip shield/drift invert system	Defense-in-Depth	
Performance of the waste package	Contribution to Performance	
Radionuclide concentration limits in water	Defense-in-Depth	Contribution to Performance
Radionuclide delay through the unsaturated zone	Defense-in-Depth	Contribution to Performance
Radionuclide delay through the saturated zone	Defense-in-Depth	Contribution to Performance
Probability of igneous intrusion		Contribution to Performance
Repository response to igneous intrusion		Contribution to Performance

The degraded and neutralized barrier analyses for the nominal scenario in Section 3 do not show significant sensitivity of expected performance to seepage. For example, increasing the seepage to the full precipitation on the mountain does not result in a significant increase in the calculated dose rate for at least 40,000 years (see Section 3.4.1). This lack of sensitivity reflects the role of the waste package and drip shield in diverting water and limiting exposure of the waste to the seepage. After 40,000 years, when both the waste packages and drip shields are degraded, the results show greater sensitivity. Consequently, seepage would contribute to system performance in the event of failure of these elements. In this sense, the seepage lends to defense-in-depth and is identified as a principal factor for this scenario.

The second principal factor in Table 4-2 is performance of the drip shield/drift invert system. This factor defines the role of the drip shield in diverting water from the waste. Further, it includes the role of the drift invert in limiting diffusive transport away from breached waste packages, while the drip shield prevents advective flow through the invert. The technical basis for the representation of drip shield performance is provided in the Waste Package Degradation PMR (CRWMS M&O 2000c). This reference summarizes the available information about potential drip shield degradation due to general corrosion (CRWMS M&O 2000c, Section 3.1.5), localized corrosion (CRWMS M&O 2000c, Section 3.1.6), stress corrosion cracking (CRWMS M&O 2000c, Section 3.1.7), and hydrogen-induced cracking (CRWMS M&O 2000c, Section 3.1.8). This PMR also discusses possible displacement of drip shield sections at installation or as a result of ground movement. The technical basis for drift invert performance is provided in the Engineered Barrier System Degradation, Flow, and Transport PMR (CRWMS M&O 2000e). This reference summarizes the information about the diffusion model for this barrier and its advective and diffusive transport properties (CRWMS M&O 2000e, Section 3.1.3).

System effectiveness and the role of this factor are shown in Figures 3-25 and 3-26. Figure 3-25 shows the effect of neutralizing the drip shield to determine its importance to base-case repository performance. Because of the waste package's robust performance, this neutralization

analysis does not show a significant contribution to system performance. However, Figure 3-26 shows other system barriers, including the drip shield, would contribute if the waste package were neutralized—complete neutralization of all waste packages results in a calculated dose rate less than 100 mrem/year. The sensitivity analysis indicates the potential drip shield/drift invert system contribution to defense-in-depth. Comparing the conservative base-case model for the drift invert with the corresponding model using a much lower diffusion coefficient shows the potential for limiting release from breached waste packages while the drip shield remains intact. Measurements by Conca and Wright (1992, p. 22) suggest the diffusion coefficient may be significantly lower than the value used in the base-case analysis, and a lower value could be demonstrated. For these reasons, the drip shield/drift invert system performance is identified as a principal factor.

The third principal factor in Table 4-2 is waste package performance. This factor defines how long the waste packages would prevent exposure of waste to water and contain the enclosed radionuclides. As a result of the selection of Alloy 22 for the outer barrier, the waste packages are calculated to remain intact much longer than 10,000 years for the range of conditions that would occur in the emplacement drifts. Accounting for uncertainties in these conditions, in waste package manufacturing, and in degradation rates, failures of the waste package are not estimated to occur within the 10,000-year regulatory period (CRWMS M&O 2000c, Section 3.2.5). The technical basis for these estimates is provided in the Waste Package Degradation PMR (CRWMS M&O 2000c). This reference describes available information regarding potential initial defects (CRWMS M&O 2000c, Section 3.1.2), general corrosion (CRWMS M&O 2000c, Section 3.1.5), phase stability and aging (CRWMS M&O 2000c, Section 3.1.4), localized corrosion (CRWMS M&O 2000c, Section 3.1.6), and stress corrosion cracking (CRWMS M&O 2000c, Section 3.1.7) of the outer waste package barrier under the full range of expected conditions.

In the current performance assessment models, waste package performance dominates the mean annual dose in the first 10,000 years. Neutralization analyses for the other barriers (see Section 3.4) show no significant effect during this period, because waste package performance is robust enough to ensure the system meets the 10,000-year performance objective, even if performance of other barriers were highly uncertain. For this reason, this factor is identified as a principal factor of the postclosure safety case.

Figure 3-26 shows that the waste package performance continues to play an important role in system performance after 10,000 years. Analyses of the juvenile failure scenario show the mean dose rate associated with the failure of a single waste package remains approximately 0.01 mrem/year or less for at least 100,000 years. The 10,000-year radiological exposure limit of NRC's proposed regulation would therefore not be exceeded until more than 2,500 such breaches were to occur. Waste packages without such a breach would continue to contain the radionuclides. Even accounting for uncertainties in the waste package degradation rate, performance of the waste package provides substantial margin in the calculated performance of the repository system.

The fourth principal factor in Table 4-2 is radionuclide concentration limits in water. This factor includes the limits for both dissolved radionuclides and those associated with colloidal suspensions. The technical basis for this factor is provided in the Waste Form Degradation PMR

(CRWMS M&O 2000d). This reference describes the solubility limits for key radionuclides (CRWMS M&O 2000d, Section 3.7.1). These include the radionuclides that dominate the estimate of mean annual dose in the first 100,000 years (i.e., technetium, iodine, neptunium, and plutonium isotopes). They also include the radionuclides whose solubility is so low in Yucca Mountain waters that they contribute negligibly to the mean annual dose (i.e., americium and curium isotopes). The effect of the solubility concentration limit is a significant contribution to repository performance. The PMR also summarizes what is known regarding the concentration limits of natural and repository-generated colloids, the factors for sorption and desorption associated with these colloids, and the concentrations associated with radiocolloids (CRWMS M&O 2000d, Section 3.8.2).

The importance of this factor to the mean annual dose is indicated in part in Figure 3-30. Although no effect is seen while the waste packages remain intact, the effect after waste packages begin to fail is significant. Increasing the solubility limits of these radionuclides so that concentrations are limited only by waste form degradation results in a large effect on the dose rate. Thus, this factor contributes to defense-in-depth to address uncertainties in waste package performance.

The importance of this factor is heightened by what is not seen in these analyses. Most radionuclides emplaced in the repository are relatively insoluble in Yucca Mountain waters and, of the hundreds of radionuclides in the waste, fewer than 30 have any potential for significant dose rate 20 km from the repository system. Most of the radionuclides are therefore so insoluble, they cannot contribute significantly to the estimate of annual dose and are not included in the calculation. The concentration limits therefore contribute significantly to performance of the system and this factor is therefore identified as a principal factor of the postclosure safety case.

The fifth and sixth principal factors in Table 4-2 are radionuclide delay through the unsaturated and saturated zones. The first of these describes the radionuclide travel through the unsaturated zone below the candidate repository and defines the minimum time before radionuclides released from the engineered barriers can reach the water table. It includes the characteristics of the unsaturated zone flow and transport system that substantially delay most of the radionuclides. The technical basis of this factor is provided in the Unsaturated Zone Flow and Transport PMR (CRWMS M&O 2000b). This reference describes water movement through the unsaturated zone below the repository (CRWMS M&O 2000b, Section 3.7.3) and potential transport of radionuclides in that water (CRWMS M&O 2000b, Sections 3.11.6 and 3.11.7). It summarizes what is known regarding both sorption and matrix diffusion properties of the unsaturated rocks below the repository (CRWMS M&O 2000b, Section 3.11.3).

The other factor, radionuclide delay through the saturated zone, describes radionuclide travel through the saturated zone below the water table and defines the minimum time before the radionuclides can reach Amargosa Valley. The technical basis for this factor is presented in the Saturated Zone Flow and Transport PMR (CRWMS M&O 2000f). This reference describes the processes that control water movement through the saturated zone and it also summarizes what is known about the transport pathways in the volcanic aquifers and the alluvium in the saturated zone (CRWMS M&O 2000f, Section 3.1.1). It summarizes what is known about the transport processes along these paths (CRWMS M&O 2000f, Section 3.2.4).

The delays that would occur through the natural barriers also contribute both implicitly and explicitly to postclosure performance. Most radionuclides in the waste are so strongly delayed in these barriers due to their sorbing affinity (CRWMS M&O 2000d, Section 3.1.1), they cannot contribute to the mean annual dose for hundreds of thousands of years. For this reason, many are excluded from the performance assessment analysis altogether. The omission of these radionuclides because of their characteristics therefore, provides an implicit contribution to the safety case. The explicit contribution of the delays for the included radionuclides is shown in Figure 3-24. Neutralization of these barrier delays separately and together results in earlier arrival at Amargosa Valley of these radionuclides. Although these shifts do not result in a large dose rate in the first 20,000 years, the small dose rate in this period exists because the engineered barriers remain intact. If these barriers did not exist, the shift would be significant. Consequently, these factors provide defense-in-depth. Because of the implicit and explicit contribution to defense-in-depth provided by these factors, they are identified as principal factors of the postclosure safety case.

Table 4-2 summarizes these findings for the nominal scenario. The table indicates that the factor that determines postclosure performance in the nominal scenario is waste package performance and that other factors provide defense-in-depth for this scenario, two of the considerations for identifying the principal factors. It is notable that none of the principal factors have been identified because uncertainties taken into account in their representation are sufficient to result in an annual dose rate exceeding the proposed radiological exposure limit. A question in this regard is the role of coupled effects introduced by heat produced by the waste. Figure 3-13 indicates that increased temperature within the emplacement drifts is not expected to have significant effects on postclosure performance (in large part because waste packages do not permit radionuclides to be released within the first few thousand years when the increase is greatest). Heat would also affect conditions outside the emplacement drifts, for example, the flow of water in the host rock. However, the long life of the waste packages would also limit sensitivity to these effects, at least for the first 10,000 years. With respect to performance in the longer term, the sensitivity analyses show some sensitivity to the flow system after about 40,000 years. However, the changes are small and the effect of thermal changes on the flow system are not likely to be sufficient to result in estimates of mean annual dose exceeding the proposed radiological exposure limit in the first 10,000 years. Consequently, these effects are not identified as a principal factor.

4.1.3 Igneous Activity Groundwater Release Scenario

Principal factors for the igneous activity groundwater release scenario are listed in Table 4-2. The igneous activity groundwater release scenario involves an intrusion of magma into the emplacement drifts with no associated eruption to the surface. The waste package and drip shield are not designed to accommodate the large changes in temperature, pressure, and chemistry that would be associated with igneous intrusion into the emplacement drifts. Consequently, in this scenario the intruding magma disrupts some of the waste packages and drip shields—the number depending upon the characteristics of the intrusion.

Waste in a disrupted waste package could be exposed to seepage into the drifts if water reaches the waste through fractures in the cooled magma body. Radionuclides in the exposed wastes could be mobilized and transported through the unsaturated zone to the water table and then to

Amargosa Valley. However, the probability of this event is very low: using the mean annual frequency (CRWMS M&O 2000h, Table 3.10.4), the probability of occurrence in 10,000 years is 1.6×10^{-4}. The mean annual dose in this case (which includes the probability of occurrence) is more than a factor of ten below NRC's proposed radiological exposure limit.

The last two factors shown in Table 4-2 are the probability of igneous intrusion and repository response to igneous intrusion (i.e., damage to waste packages and drip shields resulting from the intrusion). The technical basis for these two factors and for the estimates of system performance in the event of this scenario are presented in the Disruptive Events PMR (CRWMS M&O 2000g, Section 3.1.1). This reference summarizes what is known regarding the frequency of volcanic events, their possible impacts on the repository, and the factors affecting release of radionuclides that could occur following such an event.

Sensitivity to the probability of the igneous activity is examined in Figure 3-14. The object is to determine how strongly the mean annual dose depends on the scenario probability. The analysis shows that the mean annual dose is essentially proportional to the frequency of the event. Because this probability is a principal determinant of the mean annual dose, this factor is identified as a principal factor.

The last factor in Table 4-2 is the response of the repository to the igneous intrusion. This factor is represented by the number of waste packages and drip shields damaged during intrusion. The importance of this factor can be inferred from the results for the nominal scenario, which is also a groundwater release scenario. Those analyses show the dependence of the mean annual dose on the number of waste packages and drip shields that are failed at any time. The mean annual dose is directly proportional to this number. Accordingly, because this factor is a principal determinant of the mean annual dose for the scenario, it is identified as a principal factor.

The remaining principal factors for the igneous activity scenario in Table 4-2 are the same as the principal factors identified for the nominal scenario. Seepage into the emplacement drifts, concentration limits of the radionuclides in water, and delays through the unsaturated and saturated zones contribute directly to system performance for the igneous activity groundwater release scenario. That is, where in the nominal scenario these factors provided defense-in-depth and limit release in the event of waste package and drip shield failure, in this scenario where the engineered barriers are disrupted, these factors contribute directly. For this reason, each of these factors is identified as a principal factor for this scenario.

4.1.4 Other Scenarios

In addition to the igneous activity groundwater release scenario discussed above, a second igneous activity scenario has been considered. This scenario entails the occurrence of a volcanic eruption that results in the magma moving through the repository to the surface carrying entrained radionuclides. The eruption results in contaminated ash being transported via atmospheric dispersion to Amargosa Valley. The mean annual dose calculated for this eruptive release scenario is more than three orders of magnitude below the NRC's proposed radiological exposure. Thus, no principal factors are identified for this scenario.

NRC's proposed regulation requires evaluation of a specified scenario to assess robustness of the system in the event of inadvertent human intrusion into the repository. The following factors determine system performance in this scenario:

- Water flowing into emplacement drifts
- Radionuclide concentrations in this water
- Radionuclide delay through the saturated zone.

These factors are similar to those already included in the set for the nominal scenario. Accordingly, no new principal factors are introduced by considering this scenario.

4.2 OTHER FACTORS FOR SITE RECOMMENDATION CONSIDERATIONS

In addition to the principal factors discussed above, the current TSPA includes other factors to satisfy the DOE siting guidelines (see Appendix B). These factors include:

- Climate, infiltration, and unsaturated zone flow
- Effects of heat on the flow
- Environments in the emplacement drifts
- Environments in the waste packages
- Cladding performance
- Waste form performance
- Saturated zone flow
- Effect of wellhead on radionuclide concentrations
- Biosphere transport and uptake (biosphere dose conversion factors).

The complete set of factors takes into account all of the FEPs included in the process models. Although less important than the principal factors, the other factors listed above do play a role in the current postclosure safety case. Each of these other factors is discussed below.

4.2.1 Climate, Infiltration, and Unsaturated Zone Flow

These factors describe the ambient flow fields in the unsaturated zone, in particular, the percolation flux at the repository horizon. This flux influences the seepage into the drifts. While seepage is a principal factor, climate, infiltration, and unsaturated zone flow are not. The postclosure safety analyses in Section 3 show little sensitivity to the ambient flow fields in the unsaturated zone. Accordingly, variations of the climate, infiltration, and unsaturated zone flow at depth have little impact on the estimates of the dose rates in the current models. The current representation of the climate, infiltration, and unsaturated zone flow is described in the unsaturated zone flow and transport PMR (CRWMS M&O 2000b). This PMR describes the estimates of present and future climate and flow conditions; the calibration of current models with water potential measurements, pneumatic data, isotopic data, and temperature data; and the information on the hydrologic characteristics of the unsaturated zone (CRWMS M&O 2000b, Sections 3.5 and 3.7).

4.2.2 Effects of Heat on the Flow

This factor addresses changes in the unsaturated zone flow field due to the heat generated by the waste. These include potential thermal hydrologic changes, mineral alteration that could affect flow properties, and thermomechanical changes to fracture permeabilities. The sensitivity analyses reported in Section 3 do not show significant dependence of postclosure performance on details of the flow in the unsaturated zone. For example, Figure 3-4 shows no dependence on the flow during the first 40,000 years and very limited sensitivity for up to 100,000 years. Consequently, thermal effects that would change the flow system are not likely to be important and this factor is not identified as a principal factor. The current representation of the thermal effects on the unsaturated zone flow system is addressed in two PMRs, the Unsaturated Zone Flow and Transport PMR (CRWMS M&O 2000b, Section 3.10) and the Near Field Environment PMR (CRWMS M&O 2000i, Section 3). These PMRs describe the possible perturbations to the flow system and the relevant information from the testing and analysis program.

4.2.3 Environments in the Emplacement Drifts

The TSPA sensitivity analyses show little sensitivity to environments in the emplacement drifts in the current models. For example, the analysis shown in Figure 3-13 shows little dependence of the dose rate on the temperature within the drifts. Likewise, changes in chemistry, considered in evaluating changes in radionuclide concentration limits, as shown in Figure 3-8, show little effect. This factor does not satisfy the considerations for principal factors.

The technical basis for the representation of these environments is summarized in the Engineered Barrier System Degradation, Flow, and Transport PMR (CRWMS M&O 2000e, Sections 3.1.1 and 3.1.2). This reference describes the mechanical stresses, moisture, temperature, and chemistry on drip shield and waste package surfaces.

4.2.4 Environments in the Waste Packages

This factor describes the moisture, temperature, chemistry, and mechanical stress environments within the waste package. These environments determine the degradation of the waste form (including the cladding in the CSNF waste packages), mobilization of radionuclides, and transport of those radionuclides within the waste package. The postclosure safety analyses examining alternative designs giving different environments on the waste package (CRWMS M&O 2000h, Section 4.6) do not indicate strong sensitivity to these factors. These results also apply to the environments within the waste packages because these environments were also varied in the analyses that show limited sensitivity. Consequently, these environments have not been identified as a principal factor.

However, the current models used to represent these factors are conservative and the effects may be much more important than indicated by these models. For example, the models do not account for the effect of heat generated by the waste preventing water from seeping into the waste package. The waste packages are liable to generate enough heat to prevent liquid water from contacting the waste package for thousands of years. In the event of complete drying of the near field, water would not mobilize or transport radionuclides during this period even if waste packages and drip shields were to fail early. The effects of heat within the waste package could

have a similar effect. Temperatures sufficiently high to be above boiling would limit migration of radionuclides because water would not be available. The current models do not include credit for this effect. The benefit that would actually accrue by less conservative models is not known. Temperatures decrease with time and those in cooler parts of the repository, such as near the edge of the repository, may be below boiling well before 10,000 years and not actually provide any real benefit. No analyses are available to show a benefit at this time.

4.2.5 Cladding Performance

This factor describes the role of cladding on the spent nuclear fuel (SNF) waste form in limiting wetting of the waste, the rate of waste form dissolution, and the mobilization rate. The nominal scenario incorporates the estimate that approximately 8 percent of the cladding is breached at the time of waste package failure due to initial failures or by creep failure during the thermal period (CRWMS M&O 2000h, Section 4.1.2). The analysis also takes into account later degradation of the cladding. The effect of fully neutralizing all of the cladding shows a change of less than an order of magnitude in the calculated dose rate (see Figure 3-29). Figure 3-19 indicates the increase is about a factor of ten for CSNF waste packages, but overall effect is less when release from the (unclad) HLW is taken into account.

The only situation in which cladding would have a significant impact on performance resulting from a failure of a given waste package is when none of the cladding in that package is initially breached or degraded in the first 10,000 years. This situation would not apply to waste packages containing HLW and it would not apply to clad waste in the igneous activity scenario. Further, demonstrating that all spent fuel rods in a given CSNF waste package have no perforation of cladding is likely to be difficult. It is possible that, for selected waste packages (e.g., those with clad Navy spent nuclear fuel), it would be possible to demonstrate this to be the case; however, this demonstration could not be applied to the repository as a whole for the nominal scenario nor to the igneous activity scenario at all. Accordingly, this factor is not identified as a principal factor of the postclosure safety case.

The current representation for postclosure cladding degradation is described in the Waste Form Degradation PMR (CRWMS M&O 2000d, Section 3.4). This PMR describes the estimates of initial defects of the cladding, general and local corrosion behavior, and effects of mechanical stress on the cladding.

4.2.6 Waste Form Performance

This factor describes the rate of mobilization of radionuclides provided by degradation of the waste form (e.g., the uranium oxide ceramic matrix or HLW glass waste form) itself. In the current representation, this factor does not contribute significantly to the postclosure safety case. Analyses using the current models indicate little impact on postclosure performance from this factor and little sensitivity of the calculated dose rates to estimates of the waste form performance (CRWMS M&O 2000h, Section 5.2.4); consequently, waste form degradation is not identified as a principal factor of the postclosure safety case. The current representation for this factor is described in the Waste Form Degradation PMR (CRWMS M&O 2000d, Sections 3.3, 3.5, and 3.6). This report describes the chemistry within the waste package, degradation of the CSNF waste matrix, and the degradation of other waste forms

4.2.7 Saturated Zone Flow

This factor describes the pathways for water that could transport radionuclides in the saturated zone rocks. Groundwater flux and sorption along these paths, including their variation in the volcanic aquifers and the valley fill alluvium, are taken into account in the factor defining the delay of radionuclides through the saturated zone. The current TSPA model accounts for the flow paths in developing breakthrough curves for radionuclide migration through the saturated zone (CRWMS M&O 2000h, Section 5.2.7). Accordingly, the flow paths themselves are not identified as a principal factor. The representation of the flow pathways is described in the Saturated Zone Flow and Transport PMR (CRWMS M&O 2000f, Section 3.1 and 3.2). This report describes the regional flow system and the local flow system within that overall system. It describes features affecting the local hydrologic gradients, the system of volcanic and carbonate aquifers and the effects of faults and other features affecting the flow system.

4.2.8 Effect of Wellhead on Radionuclide Concentrations

This factor describes the radionuclide concentrations arising from dissolving or suspending the radionuclides in the water taken up by pumping of groundwater from a well. The concentrations of radionuclides are calculated by dividing the mass of radionuclides reaching the well by the volume of water used by individuals (CRWMS M&O 2000h, Section 5.2.8). Analyses of the sensitivity to this factor over the range of possibilities considered here do not show a large effect (see Figure 3-11). Because of this limited sensitivity, this factor is not identified as a principal factor.

4.2.9 Biosphere Transport and Uptake (Biosphere Dose Conversion Factors)

The annual dose calculated for long-term performance of the repository system depends upon transport in the biosphere and the uptake of the radionuclides by individuals. These effects are captured in the biosphere dose conversion factors. These factors take into account the processes associated with typical lifestyle in Amargosa Valley, including usage of groundwater. The sensitivity analyses in Section 3 (e.g., see Figure 3-12) do not show a strong dependence of the dose rate on the biosphere dose conversion factors over the range of uncertainty in this factor. Therefore, this factor is not identified as a principal factor. The technical basis for the biosphere dose conversion factors is presented in the Biosphere PMR (CRWMS M&O 2000j). This reference describes the assumptions regarding lifestyle and habits of individuals who could be exposed to radioactive material in the scenarios (CRWMS M&O 2000j, Section 3.1.2). It also describes the reference biosphere, associated pathways, and the characteristics of the critical group used in the analyses (CRWMS M&O 2000j, Section 3.1.1).

4.3 PRINCIPAL FACTORS AND WASTE ISOLATION

The results of the analyses leading to the identification of the principal barriers provide an integrated picture of the potential repository system's ability to isolate the waste. Yucca Mountain provides a physically and chemically stable rock well below the surface. As a consequence, waste can be emplaced deep underground and isolated from the surface. In the environments underground, temperature, humidity, and chemistry change slowly and permit waste packages and drip shields to maintain their integrity for tens of thousands of years. The

waste packages alone are predicted to prevent any release of radionuclides for more than 10,000 years. The drip shield and drift invert provide additional barriers to limit release of radionuclides even if waste packages fail prematurely.

A repository system at the Yucca Mountain site would, however, provide more than stable environments for the engineered barriers. It would include multiple natural and engineered barriers that could provide safety margin and defense-in-depth. The site is arid, and the combination of processes at the surface and the characteristics of the mountain limit the amount of seepage that can enter the emplacement drifts. Chemical conditions limit the concentrations of radionuclides in the water, and the combination of low seepage and these concentration limits constrains the amount of radionuclides that can be released from the engineered barriers even if they are breached.

The repository system also includes barriers to radionuclide migration away from the repository. The unsaturated zone at this site is sufficiently thick that the waste can be emplaced not only more than 200 meters below the surface, but hundreds of meters above the water table. In addition, the site is located more than 20 km from populated areas. Most radionuclides are so immobile in the rock at Yucca Mountain that they cannot migrate through this rock to populated areas. The combination of the limited ability of radionuclides to be mobilized in the engineered barrier system and the delay of radionuclides that travel away from the system limits the possible mean annual dose to levels not significantly greater than those from natural background radiation. Thus, the multiple natural and engineered barriers provide defense-in-depth and combine to restrict mean annual dose below the radiological exposure limit for more than 80,000 years.

Yucca Mountain is stable and has changed little in the last several million years. The probability of events at this site that could disrupt the repository is low. The probability of igneous activity in the next 10,000 years is less than one chance in 1,000. Analyses of this activity show that the mean annual dose arising from radionuclides that might be mobilized and carried to populated areas is low. Analyses to assess the robustness of the system with respect to inadvertent human intrusion show that in that case as well, the quantity of radionuclides that could be mobilized would be small and the amounts that could reach populated areas are almost negligible.

The combination of limited seepage into emplacement drifts, drip shield and drift invert performance, waste package performance, radionuclide concentration limits in water, and long delay of radionuclide travel through the unsaturated zone and saturated zone isolates the waste and results in mean annual doses that meet the radiological exposure limit of the postclosure performance objective. In addition, analyses of the potentially disruptive events show mean annual doses more than two orders of magnitude below this limit. The combination of all principal factors results in sufficient isolation of the waste from populated areas that the postclosure performance objective is expected to be met by a wide margin.

5. STATUS OF THE CURRENT POSTCLOSURE SAFETY CASE

The focus of the postclosure safety case is the postclosure performance objective to be established by the NRC. In its proposed rule defining this performance objective (see Appendix B), the NRC requires that demonstration that this performance objective is met include a performance assessment. Consistent with a risk-informed, performance-based philosophy, the postclosure safety case includes more than performance assessment. This case includes the following five elements:

- Performance assessment
- Safety margin and defense-in-depth
- Explicit consideration of potentially disruptive events
- Insights from natural analogues
- Performance confirmation.

NRC's proposed regulation acknowledges the value of a comprehensive safety case. The NRC emphasizes that the standard for compliance with the postclosure performance objective is reasonable assurance and indicates the value of supplementary information in meeting that standard (see Appendix B). The NRC notes, in particular, that the risk-informed, performance-based regulatory philosophy underlying this proposed regulation rests on a foundation of safety margin and defense-in-depth (NRC 1999). The postclosure safety case therefore provides these and other elements to support NRC's ability to find, with reasonable assurance, that the postclosure performance objective is met. The current status of the five elements of the postclosure safety case is summarized in the following sections.

5.1 PERFORMANCE ASSESSMENT

The performance assessment analyses conducted for the current postclosure safety case are presented in Section 4 of the report of current TSPA analyses (CRWMS M&O 2000h). Performance assessment includes TSPA analyses, that is, calculations of expected repository postclosure performance. The performance measure for these analyses is the mean annual dose to a member of the critical group, in this case a person living in Amargosa Valley. These analyses are central to the current postclosure safety case and provide an indication of postclosure safety by providing an estimate that can be compared with the radiological exposure limit of the performance objective in NRC's proposed regulation. The methodology for the TSPA analyses is summarized in Appendix C, and key results of the analyses for the current postclosure safety case are presented in Section 3.1.

To provide confidence in these analyses and to satisfy the associated NRC requirements, performance assessment includes the following:

- Supporting data for TSPA models and parameters
- An accounting of uncertainties and variabilities in parameter values
- Consideration of credible alternative conceptual models
- Technical bases for the inclusion or exclusion of FEPs
- Data regarding barrier degradation, deterioration, and alteration processes.

In addition, performance assessment includes barrier importance assessment. This assessment identifies the barriers important to waste isolation and provides a description of the capability of these barriers to isolate waste. The current postclosure safety case includes two types of barrier importance analyses: neutralized barrier analyses and degraded barrier analyses. The approach to both analyses is summarized in Appendix C.

Neutralized barrier analyses consider performance of the system without the contribution of particular barriers to limiting the movement of water or radionuclides. A neutralization analysis addresses the capability of barriers to isolate waste in two ways. First, by comparing the result of the analysis with a base case in which all of the barriers are effective, the importance of the neutralized barriers to the base-case mean annual dose can be observed. Second, by comparing the result of the analysis with a base case in which none of the barriers are effective, the contribution of other barriers can be observed.

Examples of both are shown in Figure 3-2. The lowest curve in this figure shows the estimated mean annual dose for the case where all repository barriers perform as expected. The middle curve shows the result of neutralizing the engineered barriers. The large change from the base case indicates that the engineered barriers are an important contributor to waste isolation. The upper curve in Figure 3-2 is the result where none of the natural or engineered barriers are effective and there is no dilution in the community water supply. Comparison of the middle curve with this upper curve shows that the natural barriers and dilution in the water supply, the effects not neutralized for the middle curve, are also important contributors to waste isolation.

Degraded barrier analyses do not consider full neutralization of a barrier but reduction in the ability of the barrier to limit movement of water or radionuclides consistent with the range of possible performance. These analyses explore the effects of the uncertainties in barrier performance and the contribution of this uncertainty to overall uncertainty in the estimate of mean annual dose.

Results of degraded barrier analyses for the current postclosure safety case are presented in Section 3.2, and results of neutralized barrier analyses for this case are presented in Section 3.4.

5.2 SAFETY MARGIN AND DEFENSE-IN-DEPTH

Safety margin and defense-in-depth are standard approaches to increasing confidence in the safety of a system. Safety margin means the system is designed to perform better than regulatory requirements. For example, providing margin between the radiological limit of the proposed regulation and the mean annual dose increases confidence that new information would not result in future estimates that exceed that standard. Defense-in-depth means using multiple barriers to ensure safety is not unduly dependent on any single barrier.

The adoption of a risk-informed, performance-based regulation does not impose requirements on individual barriers with respect to either margin or defense-in-depth. However, there are advantages in maintaining these concepts in the postclosure safety case. The use of safety margin and defense-in-depth addresses the fact that knowledge regarding performance of the repository system at this or any other site is necessarily imperfect. For example, heterogeneities in site properties limit complete characterization of the site. In addition, the very properties that

enhance waste isolation, such as durability of engineered materials and low permeabilities of rocks, limit the ability to obtain information. These limitations are addressed as uncertainties in conceptual models and parameter values for the probabilistic TSPA analysis.

The use of a risk-informed, performance-based approach presumes that many of the uncertainties in the FEPs important to postclosure performance can be addressed in this way. However, it is prudent to assume that not all uncertainties can be explicitly represented. Some uncertainties arise from complexities that prevent complete mechanistic understanding, validation of models, or practicable analysis. Other uncertainties are associated with unanticipated or unknown effects, such as failure mechanisms not yet revealed in spite of extensive testing. The use of safety margin and defense-in-depth is recognized as a key approach to addressing uncertainty in postclosure safety of a repository system (NRC 1999).

5.2.1 Safety Margin

The postclosure safety case addresses safety margin in two different dimensions. First, the safety case considers margin with respect to the mean annual dose in the first 10,000 years after permanent closure. The radiological exposure limit of the proposed rule defines the maximum acceptable value of the mean annual dose, and safety margin in this case is the difference between the calculated mean annual dose and this limit. Further, the safety case considers margin with respect to time. That is, the safety case considers the time after the 10,000-year period before the radiological standard might be exceeded. This margin heightens confidence that future information will not result in an unacceptable dose rate occurring in the 10,000-year period. The approach to margin in both of these senses is illustrated in Appendix C (see Section C.3).

The TSPA analyses in Section 3.1 show there is substantial margin in the mean annual dose in the first 10,000 years and in the time before the 10,000-year radiological exposure limit is exceeded. The mean annual dose in the first 10,000 years for the nominal scenario is zero, and the mean annual dose for the igneous activity scenarios is more than two orders of magnitude below the proposed radiological limit. Combining the results for these scenarios, the estimate of expected performance does not exceed the radiological limit for more than 80,000 years. That is, the repository has more than a 50,000-year margin of safety with respect to time.

5.2.2 Defense-in-Depth

Defense-in-depth is considered in two different ways in the current postclosure safety case. First, it is considered analytically in terms of barrier importance analyses. Neutralization of the waste package in the current TSPA models, as discussed in Section 3.4, results in a significant change in the mean annual dose. However, complete neutralization of all waste packages results in a peak dose rate of less than 100 mrem/year, below the average exposure from natural sources (NCRP 1987, p. 52). In addition, the analyses indicate that if a less conservative model of the drift invert diffusion coefficient were employed, complete neutralization of all waste packages would not result in a peak dose rate in the first 10,000 years exceeding 0.1 mrem/year (See Figure 3-26).

Neutralization of the other barriers individually shows much smaller changes from the base case. This result, in concert with the result for waste package neutralization, demonstrates that the system currently provides substantial defense-in-depth. Defense-in-depth is also considered qualitatively in the current postclosure safety case. The postclosure safety case accounts for barriers and other contributory aspects of the system not taken into explicit account in the quantitative estimates of postclosure performance. One such barrier is the waste package inner barrier. The current design includes a 5-cm inner barrier of nuclear grade stainless steel to provide structural strength to the waste package. This barrier delays the contact of water with the waste form, but no credit is taken for this barrier in the current TSPA analyses summarized in Section 3.1.

Other aspects that may improve performance include diffusion in the waste package, the effects of heat generated by the waste, and the protection provided by SNF cladding. Each of these aspects provides additional defense.

5.3 POTENTIALLY DISRUPTIVE EVENTS

Confidence in postclosure safety is further addressed through explicit consideration of FEPs that could lead to disruptions of the repository (see Appendix D). In general, these potential disruptions are taken into account in a systematic evaluation of the FEPs. FEPs that are not credible for a repository system at Yucca Mountain or that have sufficiently low probability of occurrence are excluded from further consideration. The criterion used for this latter determination is that the FEP has a probability of occurrence in the next 10,000 years of less than one chance in 10,000, a standard established in the proposed rule (see Appendix B) or that it can be demonstrated to contribute negligibly to the estimate of mean annual dose.

The FEPs that are not excluded are considered in the development of the process models. FEPs that are associated with process models and submodels are expressed as "process model factors." The relative importance of these process model factors and, by implication, the associated FEPs, has been evaluated in TSPA degraded barrier and neutralized barrier analyses (see Sections 3.2 and 3.4). Those that have been determined to be important to postclosure performance are included in the representations of the various scenarios accounted for in the performance assessment.

The most important of the scenarios is the nominal scenario, that is, the sequence of processes and events expected to occur over the next 10,000 years. In addition, the performance assessment addresses disruptive event scenarios. These scenarios include FEPs that have very low probability of occurrence, but warrant consideration because they have a probability of occurrence greater than 1 chance in 10,000 over 10,000 years. In particular, they include the scenarios for igneous activity at the site. They also include a scenario for inadvertent human intrusion. Because inadvertent human intrusion cannot be precluded at any site, the proposed regulation defines a representative scenario for the purpose of evaluating robustness of system performance in the event of human intrusion. All FEPs sufficiently credible to warrant consideration are explicitly taken into account in the performance assessments for these scenarios.

Some potentially disruptive events have been prominent in past technical reviews. These include:

- Igneous activity
- Seismic activity
- Future climate change
- Rise of the water table
- Repository-generated disruptions
- Inadvertent human intrusion.

The status of the evaluation of these particular events is summarized in the following sections.

5.3.1 Igneous Activity

The probability of igneous activity at the Yucca Mountain site is very low. Nevertheless, uncertainties are such that this activity cannot be precluded by current information. Consequently, the performance assessment explicitly addresses this activity. The igneous activity scenarios include magma intruding into the emplacement drifts and damaging waste packages and other engineered barriers, permitting mobilization of radionuclides. The mean annual dose associated with these scenarios does not exceed the NRC's proposed radiological limit. The analyses of these scenarios are summarized in Section 3.1.

The analyses in Section 3.1 address both groundwater release and eruptive release of radionuclides from igneous activity events that could occur. The first case includes damage resulting from these events to waste packages and drip shields that exposes the waste to water. The mean probability of this situation occurring in 10,000 years derived from the mean annual frequency in Table 3.10-5 of CRWMS M&O (2000h) is 1.6×10^{-4}, slightly greater than the probability-screening criterion (see Appendix B). Consequently, this scenario is included in the current postclosure safety case.

The second scenario is for eruptions that release waste directly to the surface. In this scenario, radionuclides reach Amargosa Valley primarily by atmospheric dispersion. The mean probability for these events occurring in 10,000 years as derived from the mean annual frequency in Table 3.10-4 of CRWMS M&O (2000h) is 0.58×10^{-4}. This mean probability is below NRC's proposed event-screening criterion of the proposed regulation, one chance in 10,000 of occurring in 10,000 years (see Appendix B). However, this scenario has been evaluated for site suitability considerations to satisfy requirements of DOE's site suitability criteria.

5.3.2 Seismic Activity

Seismic activity is common throughout the Southern Great Basin, and FEPs associated with seismic activity are included in the nominal scenario. Because of the longer period under consideration for postclosure performance than for preclosure performance, the probability of significant seismic activity is greater in the former case. The most important considerations for postclosure performance include:

- Vibratory ground motion that could shake the waste package or the drip shield
- Seismically induced rockfall that could damage the drip shield and the waste package
- Fault displacement or other ground motion that could displace the waste package or drip shield and decrease their effectiveness.

The repository system will be designed to function in spite of these effects. For example, the drip shield and waste package will each be designed to withstand the first two of these effects, and FEPs associated with them have been screened out on this basis (CRWMS M&O 2000h, Section 2.1.1). The third of these will be addressed by ensuring sufficient offset from known faults to limit displacements of the waste package or drip shield. Significant displacement on unknown faults has a probability of occurrence below the screening threshold and does not need to be considered.

5.3.3 Future Climate Change

The Yucca Mountain site has experienced changes in climate in the past, including somewhat cooler and wetter conditions that increased precipitation by about a factor of two during a glacial period about 18,000 years ago. Accordingly, the potential for climate change in the next 10,000 years is included in the nominal scenario. FEPs associated with this climate change include increased precipitation at the surface during future wetter periods, increased net infiltration into the mountain, and increased percolation down through the mountain. The potential for increased recharge of the saturated zone flow system is also included.

These effects are all taken into account in the performance assessment (CRWMS M&O 2000h, Section 3.2.1). The sensitivity analyses do not show a significant dependence of postclosure performance on the characteristics of the unsaturated zone or saturated zone flow (CRWMS M&O 2000h, Sections 5.3.1 and 5.3.7).

5.3.4 Water Table Rise

The water table could change due to increased recharge in the region. Changes in the unsaturated zone and saturated zone flow systems due to increased recharge in future wetter periods are explicitly taken into account in the current TSPA in terms of higher water table and increased groundwater flux.

FEPs for water table rise due to tectonic effects are excluded from the current TSPA. Changes in the water table of less than a centimeter occur daily and temporary changes as much as few meters have occurred as a result of seismic activity in the region. However, these changes are negligible in comparison to the distance between the repository horizon and the saturated zone. The National Academy of Sciences (National Research Council 1992, p. 55) examined available evidence for water table rise and concluded that, although future seismic events could elevate the water table, the maximum rise from this source would be less than 100 m and would be transient. Such a change is not considered to be significant and is not included in the current TSPA.

5.3.5 Repository-Generated Disruptions

The repository itself may cause changes to the near-field environment. Associated FEPs include:

- Temperature changes to the host rock caused by heat produced by the waste
- Radiolytic effects
- Pyrophoricity
- Nuclear criticality

The effects of heat produced by the waste on the hydrology, the chemistry, and the mechanical stress state are explicitly taken into account in the nominal scenario in terms of coupled effects in the near-field rock and environments within the emplacement drifts. Effects of temperature on engineered barrier material properties and on waste form degradation are also taken into account using preliminary models. The results of the analyses taking into account these effects are summarized in Section 3.1.

Pyrophoricity (potential for igniting spontaneously) has been excluded because of low probability of waste package failure and exposure of the waste. Radiolysis (decomposition of water into hydrogen and oxygen due to radiation) has been excluded because it would not affect the waste form or change waste package or drip shield degradation rates significantly. Nuclear criticality (a self-sustaining chain reaction) is of low probability because the waste package and drip shield restrict the exposure of the waste to water. Consequently, the FEPs associated with criticality have been excluded from the nominal scenario because the waste package and drip shield are expected to remain intact for more than 10,000 years (CRWMS M&O 2000k, Section 2.1.1). However, criticality remains an issue for the igneous activity scenario, because intruding magma would damage some of the waste packages, permitting water to eventually contact the waste. This situation has not yet been analyzed.

5.3.6 Inadvertent Human Intrusion

As future human activity cannot be predicted, the possibility of inadvertent human intrusion into the repository in the future cannot be precluded. Accordingly, the potential for human intrusion has been explicitly addressed in the performance assessment. The representation of the scenario for inadvertent human intrusion is specified by NRC's proposed rule (see Appendix B). The performance assessment for this scenario is summarized in Section 3.1.

5.4 NATURAL ANALOGUES

Natural analogues may provide increased confidence in the quality of the performance assessment of the repository system (see Appendix D). As defined here, natural analogues are systems that include processes similar to those that would occur in a repository at Yucca Mountain and for which information can be obtained to evaluate long-term (e.g., millennia) or large-scale (e.g., kilometers) behavior that might not be possible in laboratory and field studies. Natural analogues play two different roles in the postclosure safety case. First, they provide information that complements the laboratory and field information obtained for the Yucca Mountain site. This information includes properties (e.g., retardation factors and metal degradation rates) determined to be applicable to a Yucca Mountain repository. An example of

the use of analogues in this regard is the information regarding the initiation threshold for stress corrosion cracking in stainless steel welds that is used to estimate the threshold for welds on Alloy 22.

The second role of natural analogues is to increase confidence, through both qualitative and quantitative studies, that the current representation of the principal factors is either realistic or conservative. A number of natural analogue studies have been conducted that bear on the principal factors and these are summarized in the following sections. The set of studies is not all-inclusive. Other natural analogues are presently under consideration and would be included in the postclosure safety case if future development warrants.

5.4.1 Natural Analogue for the Unsaturated Zone Flow System

The variably saturated fractured basalt at Box Canyon, Idaho, has been used to test the approach used to model the Yucca Mountain unsaturated zone flow system (CRWMS M&O 2000b, Section 3.7.4.6). A consistent set of parameters was obtained by comparing the model predictions to measurements of infiltration-front arrival (inferred from bromide samples) at Box Canyon. The model addresses both the matrix and fracture continua of the upper basalt flow at the Box Canyon site, and parameters include the permeabilities of the fracture and matrix continua, the interfacial area between the fracture and matrix continua, and the porosity of the fracture continuum. In order to reproduce the measurements, the model required the following types of adjustments. First, the interfacial area was decreased by a factor of 100 to reduce imbibition of water from the fracture continuum into the matrix continuum during the infiltration pulses. Second, the porosity of the fracture continuum calculated using the fracture aperture inferred from pneumatic-test permeabilities was increased by a factor of 50. The arrival time of the simulated infiltration fronts is highly sensitive to the fracture-continuum porosity. The vastly different scales of Box Canyon and Yucca Mountain limit the ability to conclude that adjustments at Box Canyon also apply at Yucca Mountain. Consequently, although the Box Canyon information provides a useful general test of the modeling capability for unsaturated, fractured media, the specific site information does not yet add substantially to the other work used to calibrate and validate the Yucca Mountain unsaturated zone flow model.

5.4.2 Natural Analogues for Deposition of Calcite at Yucca Mountain

The rate of stalagmite growth in caves, tunnels, and mines around the world has been investigated for the purpose of correlating water supply rate and chemistry to calcite precipitation rate that has been used to help calibrate the unsaturated zone flow model (CRWMS M&O 2000b, Section 3.8.8). Theoretically predicted growth rates (using calcite reaction kinetics) and calculated rates (based on calcite thickness and carbon-14 ages) compare favorably.

A second analogue for calcite deposition is the calcite and opal observed in lithophysal cavities in the Topopah Spring tuff (CRWMS M&O 2000b, Section 3.9.7.1). Observations of this deposition have been used to estimate long-term seepage rates into small openings. Analysis comparing these observations with predictions indicates that (1) not all lithophysal cavities encounter seepage and (2) seepage flux is only a small fraction of percolation flux. Both conclusions corroborate the conceptual model of the unsaturated zone flow system. In particular, they support the concept of a capillary barrier limiting seepage into small openings.

5.4.3 Natural Analogues for Seepage in Emplacement Drifts

Rainier Mesa, located approximately 30 km northeast of Yucca Mountain, is underlain by a sequence of welded and nonwelded tuffs similar to those that underlie Yucca Mountain. Current precipitation at Rainier Mesa is approximately twice that at Yucca Mountain, and seepage into tunnels at Rainier Mesa is considered to be an analogue for seepage into emplacement drifts at Yucca Mountain under future wetter conditions (CRWMS M&O 2000b, Section 3.9.7.2). Percolation flux is estimated to be about 25 mm/year at Rainier Mesa, and joints, including throughgoing faults, intersecting tunnels have yielded water. The seeping features are thought to be pathways for flow from a perched water zone. The seepage is geochemically similar to meteoric water and is associated with fast flow pathways. Seepage at Rainier Mesa occurs only in tunnels constructed in the zeolitic tuff units: no seepage is observed in the vitric tuff units. These observations suggest that seepage into tunnels may be localized and restricted to certain flow paths and units. This picture is consistent with the model for seepage at Yucca Mountain under future wetter conditions.

Another analogue for seepage into emplacement drifts at Yucca Mountain is seepage into natural caves at Altamira, Spain (CRWMS M&O 2000b, Section 3.9.7.3). These caves are located in the unsaturated zone of fractured limestone bedrock with clay-rich layers. Precipitation is approximately ten times higher than at Yucca Mountain; nevertheless, seepage rates into these caves are very small. The low rates corroborate the low rates of seepage predicted for emplacement drifts at Yucca Mountain.

A third analogue for seepage into the emplacement drifts is the seepage into sealed drift segments in the Exploratory Studies Facility at Yucca Mountain (CRWMS M&O 2000b, Section 3.9.7.4). No natural seepage into the Exploratory Studies Facility has been observed. However, evaporation and moisture removal by ventilation could be a factor. Some drift segments in the Exploratory Studies Facility have been sealed by bulkheads to isolate them from ventilation, and no seepage has been observed so far in them. The absence of visible seepage into these segments corroborates the modeling results that suggest current percolation fluxes are smaller than prevailing seepage thresholds.

5.4.4 Natural Analogues for Thermal-Chemical-Hydrologic Effects

An analogy for coupled effects associated with a potential repository at Yucca Mountain is the fossil hydrothermal system at this site (CRWMS M&O 2000b, Section 3.10.9). After formation of the major zeolitic horizons about 12 million years ago, deep-seated hydrothermal activity persisted for more than a million years. This activity was limited to temperatures of about 90-100°C, comparable to those that could occur in portions of the host rock in some locations. Measured temperatures in the repository block are consistent with inferred paleotemperatures. Further observed mineral abundances are consistent with model estimates.

An additional analogue for these processes is provided by the contact between igneous intrusions and hot rock at the Banco Bonito obsidian flow near the Valles Caldera, New Mexico (CRWMS M&O 2000b, Section 3.10.9). The obsidian, initially at temperatures of 850°C, heated tuff in the canyon walls to 150°-350°C for decades, vaporized pore water, and caused refluxing of water.

No evidence of hydrothermal alteration has been noted. Overall, the effects of this heating on the unsaturated environment appear to be slight and limited to the tuff nearest the contact.

Another analogue with similar effects is that at Grants Ridge, New Mexico (CRWMS M&O 2000b, Section 3.10.9). Despite the high temperature associated with the basaltic intrusion at this site (> 700°C), there is no evidence of pervasive hydrothermal circulation or alteration of country rock.

A fourth analogue is found at the Paiute Ridge area of the Nevada Test Site (CRWMS M&O 2000b, Section 3.10.9). In this case, late Miocene basaltic magma intruded a sequence of tuffs in a single magmatic pulse as dikes, sills, and lopoliths. Alteration of the rock occurred within 60 m of the intrusion, including filling of pores with silica at fracture-matrix interfaces. However, the fractures themselves are observed to remain unfilled, contrary to predictions of fracture filling in the vicinity of a heated repository. However, in this case, as in the others, the effects of the high temperature on unsaturated environments have been slight. These field studies provide natural analogues for the effects of heat that would be generated by the waste.

5.4.5 Natural Analogue for Radionuclide Delay through the Unsaturated Zone

Processes that formed the Nopal I uranium deposit at Peña Blanca, Chihuahua, Mexico (CRWMS M&O 2000b, Section 3.11.12.1) may be a natural analogue of those at Yucca Mountain. From uranium-thorium dating, it appears that transport of uranium in the fractures occurred more than 300,000 years ago. Subsequently there has not been significant uranium-238 or uranium-235 redistribution; uranium, thorium, and protactinium isotopes in fracture filling minerals have apparently survived hydrological disturbance from surface water infiltration. Recent data obtained from thermal ionization mass spectrometry indicate that the geochemical system at the Nopal I site restricts actinide mobility in the unsaturated environment. By analogy, the tuffs at Yucca Mountain should have similar properties. This information complements laboratory and site data regarding transport characteristics of the Yucca Mountain site.

5.4.6 Natural Analogues for Radionuclide Delay through the Saturated Zone

A number of sites bearing on saturated zone flow and transport have been studied as natural analogues for the Yucca Mountain site (CRWMS M&O 2000f, Section 3.4.5.2). These include studies at Alligator Rivers in the Northern Territory of Australia, the Poços de Caldas caldera in Brazil, the Oklo uranium mine in the Republic of Gabon, the Palmottu uranium-thorium deposit in southwestern Finland, and the Cigar Lake uranium deposit in northern Saskatchewan, Canada.

The Alligator Rivers Analogue Project investigated the migration of radionuclides in saturated media from the Koongarra secondary enriched-uranium deposit (CRWMS M&O 2000f, Section 3.4.5.2.1). Dissolved uranium is transported from the deposit to a succession of identifiable zones. The migration of the uranium involves dispersion, sorption, and other processes like those that would apply at the Yucca Mountain site. The Alligator Rivers site differs in several important ways from the Yucca Mountain site; in particular, the transport pathways are close to near-surface oxidizing conditions and subject to monsoonal climate. Nevertheless, the dissolution and reprecipitation of uranium in the dispersion plume provide

valuable information that can be used in calibrating the saturated zone flow and transport model used for the Yucca Mountain site.

The Poços de Caldas caldera has been the focus of a natural analogue study involving the Osamu Utsumi and the Morro do Ferro uranium mines (CRWMS M&O 2000f, Section 3.4.5.2.2). The Osamu Utsumi mine is a uranium ore body with subsidiary thorium, zirconium, and rare earth elements, and the groundwaters there have high concentrations of uranium. The Morro do Ferro mine is a thorium and rare earth element ore body with subsidiary uranium, and the groundwaters there contain both uranium and traces of thorium. Studies have evaluated transport of both dissolved constituents and colloidally transported radionuclides. The results of the colloid study suggest that transport of radionuclides and other trace elements by colloids in deep groundwaters in this area do not play an important role in the process of leaching and erosion of the ore deposits. The studies point to the filtration of these colloids as an efficient process in limiting its effect. These results provide information that is useful in evaluating the model for transport of dissolved and colloidally transported radionuclides in the saturated zone at Yucca Mountain.

The Oklo uranium mine contains the only known examples of a natural fission reactor and has produced transuranic elements and fission products in the natural environment like those that would be in the wastes at a potential Yucca Mountain repository (CRWMS M&O 2000f, Section 3.4.5.2.3). Approximately two billion years ago, the natural reactors were active for about a million years and produced uranium-234, plutonium-239 and 242, and other radionuclides including possibly neptunium-237. Measurements indicate the uranium and plutonium isotopes have not migrated from the core uraninites since the reaction ended about 2 billion years ago. Analyses of ruthenium-99 in clay deposits some distance from the natural reactors suggest technetium-99 may have migrated to these clays. No trace of mobile fission products has been found, suggesting migration to larger distances for these radionuclides. These results are consistent with the model predictions for the Yucca Mountain and suggest the overall consistency of the model for radionuclide transport in saturated rocks.

The Palmottu uranium-thorium deposit provides additional natural analogue information regarding the saturated zone flow and transport model (CRWMS M&O 2000f, Section 3.4.5.2.4). The deposit is hosted in Precambrian gneisses and is up to 15-m thick, 100-m long, and extends to depths of 300 m. The ongoing natural analogue study has concentrated on colloidal transport and matrix diffusion in fractured crystalline rocks, processes potentially important in the fractured tuffs at the Yucca Mountain site. Measurements show greater concentration of dissolved uranium in oxidizing upper waters than in the deeper, more reducing waters, even though the concentration of uranium in the rock is greatest at depth. The opposite trend is observed with uranium associated with colloids in the water. These results suggests more limited transport of the colloidally associated radionuclides than the dissolved radionuclides, consistent with the current Yucca Mountain model

The Cigar Lake uranium deposit has been studied as a natural analogue for radionuclide transport under saturated, reducing conditions deep below the water table (CRWMS M&O 2000f, Section 3.4.5.2.5). The deposit lies at the contact between crystalline rocks of the Precambrian shield and the overlying saturated sandstones; however, the system is still considered to have applicability to the Yucca Mountain site. Uranium has migrated upward away from the deposit,

but naturally produced plutonium and technetium have migrated little and remain within the deposit. Studies show that half or more of the uranium in the water is carried colloidally. Radiolysis of the groundwater within the ore body may be the cause of locally oxidizing conditions and concomitant formation of iron-oxyhydroxide colloids at the top of the ore body. Migration of uranium-bearing colloids is thought to be the cause of depletion of the ore body and the concentrations of uranium in water above the body. Analyses indicate colloid filtration by clay minerals has played a role in limiting the uranium concentrations in these waters.

5.4.7 Natural Analogue for Performance of the Waste Package and the Drip Shield

The Akrotiri archeological site at Santorini, Greece (CRWMS M&O 2000b, Section 3.11.12.3) is similar to Yucca Mountain in its silicic volcanic rocks, dry climate, and unsaturated subsurface conditions. It provides a natural analogue for stability of metals in underground conditions. Bronze and lead artifacts were buried under 1.5 to 2 m of volcanic ash 3,600 years ago by the Minoan eruption. The possibility of migration of copper, tin, and lead was investigated by selective leaching of packed earth and bedrock samples collected directly beneath the site where bronze and lead artifacts were excavated. Field data indicate that little of the bronze material has been transported from its primary location. The total amount of copper removed from the artifacts was less than 0.1 percent of the volume of the artifacts. Neither copper nor lead is detected more than a meter away from the buried artifacts. Although these metals are not the same as those considered for the waste package or drip shield, the Akrotiri study suggests that metals could be preserved in the ground for a long period even in an oxidizing environment such as that at Yucca Mountain.

5.4.8 Natural Analogue for Waste Form Degradation

The Nopal I uranium deposit at Peña Blanca, Chihuahua, Mexico, also provides a natural analogue of the alteration of the uranium oxide spent nuclear fuel (CRWMS M&O 2000d, Section 3.3.2.3). Uranium dioxide occurs naturally in nature as the mineral uraninite, and the Nopal I site contains substantial quantities of uraninite in geologic, geochemical, and hydrogeologic environments similar to those at Yucca Mountain. The mineral phase assemblage at Nopal I is similar to that observed experimentally in CSNF laboratory tests. The general agreement between the observed alteration products in the natural analogue and in the experimental tests provides confidence that the mechanisms of spent fuel degradation are understood and that the forward dissolution model used to represent CSNF degradation in the performance assessments provides a bound for the CSNF degradation rate.

5.4.9 Natural Analogue for Igneous Activity Eruptive Processes

Cerro Negro is a basaltic, cinder-cone volcano located 20 km northeast of the city of Leon in northwestern Nicaragua that provides a natural analogue for such an event at Yucca Mountain (CRWMS M&O 2000h, Appendix C). Following a volcanic eruption with a significant ash plume at Cerro Negro in November 1995, the thickness of the deposited volcanic ash was measured immediately after the eruption. These measurements have been compared with the results of modeling using the same methodology and computer model developed for the Yucca Mountain site. The calculations compare well with the observed data for distances from the volcanic vent greater than 10 km. For distances less than 10 km, the calculation gives ash

thicknesses greater than the observed data. This comparison suggests the methodology is likely to provide a reasonably conservative approach to analyzing the igneous activity eruptive scenario at Yucca Mountain.

5.5 PERFORMANCE CONFIRMATION

The license application will include a commitment to conduct a long-term testing program that is consistent with the requirements of the NRC's proposed regulation (see Appendix B). This testing is designed to confirm that barriers identified in the license application as important to waste isolation are performing as expected and that key parameters fall into the ranges specified in the license application. This testing is dictated by two considerations. First, it is required by specific language of the proposed regulation. Subpart F of NRC's proposed regulation includes specific guidance regarding the testing program (see Appendix B). Second, the testing must address the principal factors, those factors critical to meeting the postclosure performance objective. The testing will focus on those aspects of these factors that can be practically addressed in a testing program following the submittal of the license application.

Results of the testing would have implications for decisions that would be made after license application. The results would provide information to support the decision-making associated with permanent closure of the repository. In addition, the significance of results obtained during performance confirmation regarding unanticipated performance with respect to a principal factor would be evaluated, and one potential action that could be recommended is retrieval.

A generic Performance Confirmation Plan was written to provide general understanding of the requirements for the performance confirmation program (CRWMS M&O 1997). This plan was revised (CRWMS M&O 2000l) to provide more specific information with regard to the requirements of NRC's proposed rule (see Appendix B). The revised plan addresses the preliminary principal factors identified in Revision 3 of the Repository Safety Strategy (see Section 3.1 of CRWMS M&O 2000m).

The current Performance Confirmation Plan does not address several important areas. First, the final rule has not yet been promulgated, and the plan would need to be updated to address the requirements of NRC's final regulation. Second, the plan does not yet address the final set of principal factors of the postclosure safety case. Third, the results of performance confirmation testing and analyses have implications for other programs including those related to potential retrieval and permanent closure. The details of the relationship between these results and decision-making in those areas have not yet been defined.

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6. UNCERTAINTY IN THE POSTCLOSURE SAFETY CASE

The central challenge to confidence in postclosure safety of a repository system is uncertainty arising from complexity in the system and the need to represent performance of the system far into the future. For this reason, treatment of uncertainty is essential to the postclosure safety case. The general approach that has been adopted to address this uncertainty is summarized in Appendix D. This section summarizes the application of this general approach in the current postclosure safety case and the plans for addressing remaining issues to complete the case for the site recommendation and licensing

6.1 TREATMENT OF UNCERTAINTY IN THE CURRENT POSTCLOSURE SAFETY CASE

The approach to uncertainty taken for the postclosure safety case follows the method outlined in the Nuclear Energy Agency paper, *Confidence in the Long-term Safety of Deep Geological Repositories—Its Development and Communication*, (Nuclear Energy Agency 1999). Chapter 4 of this reference points at two categories of measures to address uncertainty:

- Robustness of the system concept
- Quantification of uncertainties.

The following sections summarize the actions taken in each of these areas so far.

6.1.1 Robustness of the System Concept

Robustness contributes to confidence through measures taken to reduce uncertainties or to mitigate their effects. The current postclosure safety case includes the following such measures:

- Safety margin
- Defense-in-depth
- Flexible repository design to address uncertainties associated with thermal effects.

Safety Margin—The current system concept includes substantial safety margin. No releases are expected to occur under nominally expected conditions because the waste package is projected to remain intact for much longer than 10,000 years under these conditions. Even considering potential disruption of waste packages in an igneous intrusion event, the repository system is expected to meet the proposed radiological exposure limit by more than two orders of magnitude in the first 10,000 years and to maintain the mean annual dose below this limit for at least 80,000years. Consequently, future changes in information that might affect the mean annual dose would have to be significant to cause the limit to be exceeded.

Defense-in-Depth—Defense-in-depth means using multiple barriers to ensure safety is not unduly dependent on any single barrier and that failure of any one barrier does not result in failure of the entire system to meet its performance objective. Analyses assuming failure of the waste package show that the other natural and engineered barriers perform well enough to reduce the mean annual dose by more than six orders of magnitude. The analyses show that without credit at all for the waste packages, the remaining barriers perform well to result in a mean

annual dose below 100 mrem/year. Therefore, the current system concept provides a substantial degree of defense-in-depth.

Flexible Design to Address Uncertainties Associated with Thermal Effects—Heat introduces uncertainty into the system because the system must operate under conditions different from the ambient conditions observed in the natural system. While heater tests have provided important information about some of the effects of increased temperatures in the rock, these tests are necessarily of a smaller scale than the potential repository. Consequently, uncertainties still remain in spite of these tests. These uncertainties are approached in part by developing a robust design whose performance is relatively insensitive to near-field thermal effects. Further, the present repository design is flexible, maintaining the option for either above-boiling or lower-temperature operation during the thermal period. Thus, the current system concept permits selection of the option after additional testing and analyses have been conducted to address these uncertainties.

6.1.2 Quantification of Uncertainties

In addition to robustness of the system concept, confidence is enhanced by identifying, quantifying, and explicitly addressing the uncertainties in the performance assessment analyses. The probabilistic approach to performance assessment (see Appendix C) in the current postclosure safety case permits explicit treatment of uncertainties. Uncertainty and variability in process models can be represented in terms of probability distributions for the parameters of those models. These probability distributions are sampled in a Monte Carlo approach to produce a probability distribution for the estimate of the annual dose. The variance around the mean annual dose arising from the uncertainties in the process models is therefore calculated directly by quantifying those uncertainties.

In principle, all such uncertainties can be quantified either in terms of probability distributions developed from variability in measured properties, estimates of this variability, or judgments by experts in the appropriate fields. In practice, some models are sufficiently complex and knowledge is sufficiently limited that conservative or bounding parameters are used rather than attempting to develop explicitly quantified uncertainty estimates. Nevertheless, quantification of uncertainty is extensive in the current TSPA and analyses using this quantification are reported in Section 5 of CRWMS M&O (2000h). The resulting effect of the quantified uncertainties is shown in "horsetail" diagrams, i.e., calculations of the annual dose for all of the Monte Carlo realizations. The mean annual dose and its variance are explicitly developed from the set of realizations.

In addition to the estimate of the overall uncertainty, the effects of uncertainties in particular process model components are studied through uncertainty importance analyses. These include degraded barrier analyses, such as those presented in Section 3.2, in which a particular uncertain parameter is fixed at an extreme value (i.e., the 5th or 95th percentile of the probability distribution) to observe the effect on the mean. They also include more comprehensive analyses in which the contribution of the variance of individual parameters to the variance of the overall annual dose is estimated formally.

The DOE is currently reviewing the treatment of uncertainties within the TSPA. The main goal of this review is to support integration and communication of the treatment of the uncertainties. The review is considering the way quantified uncertainties in the individual models are propagated through TSPA to the estimate of annual dose. The review will also consider the approach to unquantified uncertainties, including the use of conservatism, bounding values, and mitigation techniques.

The DOE is also reviewing the conservatism in the current TSPA. The use of conservatism to address uncertainties in models is generally considered to lead to increased confidence in the postclosure safety case. However, confidence may not actually be increased if the degree of conservatism is not understood or defensible or if the conservatism limits understanding of the system. In particular, an understanding of the contribution of some barriers to the postclosure performance may be obscured by conservative representations for other barriers.

The DOE's review of TSPA model conservatism is focusing on the following:

- Identification of the importance of known model approximations to performance estimates
- Estimation of reasonably expected probability distributions for model parameters
- Calculation of expected performance with the reasonably expected probability distributions
- Comparison of the results with the estimates using the model approximations.

The goal is to provide an assessment of whether the calculated mean dose rates are, on balance, conservative and of the approximate magnitude of that conservatism. The review is addressing both short-term (10,000 years) and longer-term (peak) dose rate estimates.

6.2 ISSUES REQUIRING FURTHER ATTENTION

While the current postclosure safety case addresses uncertainties, some issues remain and need to be addressed in completing the postclosure safety case for site recommendation and licensing considerations. Resolution of the following issues would be accomplished by the performance of work addressing the various elements of the postclosure safety case.

6.2.1 Role of the Natural Barriers

The performance objective of NRC's proposed regulation specifies that the repository system include multiple barriers, including both natural barriers and an engineered barrier system (see Appendix B). However, the current postclosure safety case in the nominal scenario is dominated by performance of the engineered barriers. In this case, uncertainties common to all engineered barriers (e.g., experience limited to a few years or decades) limit confidence in estimates of system performance over thousands of years.

The contribution of the natural barriers is examined in two ways in the current postclosure safety case. First, the postclosure safety case identifies features of the natural system that are important factors in waste isolation. These features include, for example, the following:

- Existence of the mountain for millions of years
- No significant erosion or tectonic activity
- Physically and chemically stable host rock
- Insulation of waste at depth from surface conditions
- Environments at depth that support long-lived engineered barriers
- Ability to locate the waste far from populated areas.

These features of the natural system are addressed qualitatively in the postclosure safety case. They are not treated explicitly in the current TSPA analyses; consequently, their contribution to meeting the postclosure performance objective cannot be quantified.

The second approach to natural barriers addresses this limitation. Degraded barrier and neutralized barrier analyses in which performance of the engineered barriers is diminished have been performed to examine potential contributions of the natural barriers (see Section 3). For example, analyses without any contribution from waste package performance show a mean annual dose of about 100 mrem/year. This result represents a reduction in the mean annual dose by more than six orders of magnitude (see Figure 3-2).

These analyses suggest the contribution of the natural barriers is substantial. However, additional development in the models for the performance of these barriers might show greater contribution to overall system performance from the natural barriers. If so, confidence in the postclosure safety case could be increased. Possible candidates for model development are identified in the following section in the context of increased contribution to defense-in-depth.

6.2.2 Reliance on Waste Package Performance

As already pointed out, the current TSPA results for the nominal scenario depend strongly on the performance of the waste package. This effect has two implications for confidence in the postclosure safety case. First, uncertainties in the representation of waste package performance will be extremely important to the postclosure safety case for the site recommendation and licensing considerations. Of particular importance in this regard is the current waste package degradation model. The current model is based on two years of project data and a few decades of data from other sources; consequently, extrapolation to performance for 10,000 years is a challenge. While progress in mechanistic understanding that would permit long-term predictions has been significant, uncertainty remains in several areas. One of these pertains to the chemistry of the water contacting the waste package. Limited flowing water at depth in Yucca Mountain increases the difficulty in determining trace constituents of this water that might affect localized corrosion of Alloy 22. In addition, dust and materials introduced into the repository could affect the water chemistry important to the corrosion rates. Other areas of importance include stability of the passive oxide film and its growth kinetics and the effects of phase stability and microbial effects on localized corrosion rates. Increased information in these areas is important to increased confidence in the postclosure safety case.

The second implication of the current TSPA results is the value of increased contribution from barriers other than the waste package. Confidence in postclosure safety would be increased if it could be shown that these other barriers contribute to defense-in-depth. The neutralization analyses show significant performance even when all waste packages are assumed to fail in the first 100 years after emplacement (see Section 3.4.2). There is no requirement for the system absent the waste package to meet the proposed radiological exposure limits; however, demonstration that the requirement could be met without heavy reliance on the waste package would increase confidence in the postclosure safety case.

The current information indicates that the case for defense-in-depth could be enhanced. For example, the analyses showing the sensitivity to the drift invert diffusion coefficient (see Figure 3-25 and associated discussion in Section 3.4) indicate a substantial impact by using a less conservative model for diffusion through the drift invert. The conservative model for radionuclide transport out of the engineered barrier system was chosen to simplify the defensibility of that model and to reduce the need for additional measurements. In view of the potential impact of reduced conservatism in this model, as indicated by the results for a lower diffusion coefficient, future evaluation of the engineered barrier system transport model appears to be warranted.

Measurements of the drift invert diffusion coefficient may not be the only way to evaluate the potential for increased defense-in-depth. Conservatism chosen for other models in order to simplify their technical bases could be limiting the amount of defense-in-depth evident in the analyses. In particular, conservatism in the following models may warrant evaluation:

- Seepage
- Emplacement drift environments
- In-package transport and transport away from the waste package
- Drift shadow
- Unsaturated and saturated zone transport.

Seepage—The model for seepage into emplacement drifts used in the performance assessments takes into account threshold effects—that is, reduction of seepage to a zero value at low percolation flux. However, if additional measurements could demonstrate a higher seepage threshold (e.g., due to thermal effects) than considered in the current model, this information could provide additional confidence and reduce the burden on waste package performance.

Emplacement drift environments—Likewise, the current model for the drift environment does not take credit for limiting contact of water with the waste, e.g., by evaporation of water before it can contact the waste. Further, the model does not take into account the effect of long-term passive ventilation in reducing the amount of moisture in the emplacement drifts. Additional measurements and analyses of the drift environments could provide additional confidence in the safety case.

In-package transport and transport away from the waste package—The effects of heat in preventing flow of water into breached waste packages and transport of radionuclides out of them and to the invert are neglected in the current model because the waste packages remain

intact throughout the thermal period. However, if these effects could be shown to be effective, reliance on the waste package remaining intact for this period could be reduced.

Drift shadow—The current analyses do not take into account reduction in moisture in the rock directly below the emplacement drift due to the presence of that drift. Because of capillary effects that restrict seepage into the drifts, some of the percolation flux is diverted around the drifts, creating an effective drift shadow below them. The reduction of moisture in this zone may restrict advective and diffusive transport of radionuclides there.

Unsaturated and saturated zone transport—Similarly, the models for transport of radionuclides in the unsaturated and saturated zones could provide additional defense-in-depth if these could be shown to provide more delay in radionuclide movement than in the current models. Areas that could be evaluated include the following:

- Conservative estimates of transport through fractures in the rock below the repository
- Conservatively low estimate of the effect of matrix diffusion on radionuclide transport in the saturated zone
- Conservative estimate of radionuclide pathways through alluvium in the saturated zone
- Dilution of radionuclide concentrations by mixing of waters upwelling from the carbonate aquifer
- Conservative estimate of sorption in the alluvium.

6.2.3 Repository-Generated Heat and Coupled Effects

Heat produced by radioactive decay of the waste, whether the repository is operated in an above-boiling or lower-temperature range, would add uncertainty to estimates of postclosure performance. The higher the temperature, the greater the disparity from the range explored during site characterization. For example, the greater the heat, the greater the likelihood that water in the near-field rock would be mobilized and moved to regions that have not been fully characterized.

Uncertainties due to heat generated by the waste are addressed in part through a robust waste package whose performance is expected to be relatively insensitive to near-field thermal effects. However, the need to assure contribution of other barriers and prevent undue reliance on the waste package requires attention to uncertainties associated with thermal effects.

The primary concerns are related to coupled processes. These include thermal effects on the distribution and flow patterns of water in the host rock, thermal effects on the mineralogy affecting both flow of water and transport of radionuclides, and thermally driven changes to mechanical properties of the rock. Additional work is needed to address these coupled processes for the postclosure safety case for the site recommendation and licensing considerations.

A related issue is the approach to evaluating design options associated with the thermal strategy for the repository system. The present repository design is flexible, maintaining the option for

either above-boiling or lower-temperature operation during the thermal period. The general plan is to perform additional testing and analyses to determine which option should be implemented.

6.2.4 Potential for Igneous Activity

Igneous activity presents several issues for the postclosure safety case. Risk estimates for igneous activity scenarios are very low (see Section 3.1.2). Uncertainties taken into account in making these estimates do not show a significant likelihood the proposed radiological exposure limit would be exceeded. Nevertheless, the possibility of igneous activity in the next 10,000 years presents perceptual problems that affect confidence in the postclosure safety case.

One issue centers on the igneous intrusion eruptive event. The current estimate of the mean probability of this event is less than one chance in 10,000 of occurring in 10,000 years. This event is considered in the current TSPA because of requirements to address this event in DOE's siting guidelines (see Appendix B). However, in its risk-informed approach to focusing on the areas of greatest importance to public risk, the NRC proposes that only events that have at least one chance in 10,000 of occurring over 10,000 years be considered (see Appendix B). Consequently, if evidence continues to support a conclusion that the mean probability of occurrence of this event in 10,000 years is less than one chance in 10,000, this scenario would not be included in the postclosure safety case for a license application.

A second issue is the probability weighting associated with the estimate of risk or expected performance. In this approach, the importance of uncertainty in the estimates of consequences of an igneous intrusion is diminished by the probability weighting. There may be value, therefore, in presenting the effects of these uncertainties conditioned on the occurrence of the event, i.e., in the absence of probability weighting, in the postclosure safety case.

6.2.5 Repository Performance beyond the Regulatory Period

The proposed NRC regulation specifically addresses only the first 10,000 years after repository closure (see Appendix B). However, there is value in demonstrating that the regulatory limits are met for some presently undefined period of time beyond 10,000 years. One way to deal with this issue would be to identify the model simplifications that have been made to provide conservatism in the estimate of 10,000-year performance and to evaluate the impact of these simplifications on longer-term performance. Simplifications in the estimate of the amount of water contacting the waste and the solubility of key radionuclides, notably neptunium-237, may be resulting in conservative estimates of the long-term dose rate.

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7. COMPLETING THE POSTCLOSURE SAFETY CASE FOR SITE RECOMMENDATION AND LICENSING CONSIDERATIONS

The postclosure safety case to support site recommendation and licensing considerations is nearly complete. However, a few issues remain. Work needed to address these issues is described in Section 6 and summarized in Table 7-1. This table also summarizes additional work needed to complete the postclosure safety case, such as completion of the ongoing uncertainty review and the Performance Confirmation Plan.

Table 7-1. Plans for Work to Complete the Postclosure Safety Case for Site Recommendation and Licensing Considerations

Element of Postclosure Safety Case	Issues Remaining in the Postclosure Safety Case	Work Needed to Address Remaining Issues
Performance Assessment	• Management of uncertainties	• Complete the review of treatment of uncertainty and conservatism in TSPA models
	• Reliance on waste package performance	• Conduct additional testing and analysis of waste package degradation
	• Repository-generated heat and coupled effects	• Assess coupled effects • Refine models and assess lower-temperature repository design options
Safety Margin and Defense-in-Depth	• Role of natural barriers • Reliance on waste package performance	• Evaluate potential for drip shield/drift invert system to provide increased defense-in-depth • Evaluate other process models to determine potential for increased margin and defense-in-depth – Seepage – Emplacement drift environments – In-package transport and transport away from waste package – Drift shadow – Unsaturated and saturated zone transport
Potentially Disruptive Events	• Potential for igneous activity at Yucca Mountain	• Provide conditional results for igneous activity groundwater release scenario
Natural Analogues	• Management of uncertainties	• Evaluate value of studies of natural analogues after license application
Performance Confirmation	• Performance confirmation program	• Complete Performance Confirmation Plan for a License Application

Table 7-1 does not identify all the work that would be done to complete the postclosure safety case. The areas specified in this table include evaluation of current models and possible model development, if warranted, to address areas of uncertainty and conservatism. As evaluations are conducted in these areas, additional areas may be identified by the DOE that could merit attention. In addition, future work will take into account possible enhancements to repository system design and new site information. In particular, if the enhancements or new information change the understanding of what is important to postclosure safety, the plans for future work will be revised as appropriate. Finally, work will also be conducted as a consequence of agreements with the NRC regarding key technical issues identified in interactions between DOE and NRC. Work in these areas will be defined later. The following sections address the specific areas of work identified in Table 7-1.

7.1 PLANS FOR PERFORMANCE ASSESSMENT

The plans for performance assessment include preparation of the final performance assessments for the site recommendation and licensing considerations. These include updating the TSPA models with any new information, completing TSPA analyses with the updated models, conducting sensitivity and uncertainty analyses, and completing barrier importance analyses.

In addition, the plans include the work to address the remaining issues. The work supporting these efforts is summarized in Table 7-1. This work includes:

- Complete the review of the treatment of uncertainty and conservatism in TSPA models
- Conduct additional testing and analysis of waste package degradation
- Address coupled effects
- Refine models and assess lower-temperature repository design options

7.1.1 Complete the Review of the Treatment of Uncertainty and Conservatism in TSPA

The review of the treatment of uncertainty and conservatism in TSPA will help in the integration and communication of the uncertainty. The goals of this work are the following:

- Evaluate the degree of conservatism used in the TSPA
- Develop alternative representations of key unquantified uncertainties and evaluate the impact of these changes on the TSPA results
- Consider the uncertainty associated with alternative models or assumptions.

7.1.2 Conduct Additional Testing and Analysis of Waste Package Degradation

The analyses in Section 3 show the important role of waste package performance to the postclosure safety case. The TSPA model for this principal factor must have an adequate technical basis, and additional work is needed. This work includes continued testing to provide longer-term corrosion data and additional information on stress corrosion cracking and stress mitigation effectiveness.

Work is needed first to improve the model for the chemistry that would contact the waste packages. Second, work is needed to improve mechanistic understanding of general corrosion, including the nature and stability of the passive oxide layers. Third, work is needed to improve understanding of time dependence of corrosion and to conduct longer-term testing in this regard.

7.1.3 Assess Coupled Effects

The object of this work is to improve the robustness of analyses of thermal effects that lead to uncertainties in the distribution and flow patterns of water and changes in mineralogy and hydrologic properties due to chemical and mechanical couplings. The tests and analyses would therefore provide a basis for assessing the appropriateness of the current TSPA models for coupled processes.

The primary additional testing will be provided through in situ thermal testing and comparison with laboratory experiments on water movement in a thermally perturbed medium. Additional modeling exercises will be useful in helping to quantify uncertainties related to thermal-hydrological-chemical and thermal-hydrological-mechanical issues. In addition, documentation will be provided describing how the various coupled process models are integrated and implemented and how the results are used in TSPA.

7.1.4 Refine Models and Assess Lower-Temperature Repository Designs

Finally, the TSPA needs to be expanded to address the full range of design options. The current TSPA analyses focus on a single design option. However, the current design encompasses a range of thermal loading strategies. Analyses are therefore needed to address the full range of thermal loading strategies. In particular, since both above-boiling and lower-temperature options are maintained in the current design, analyses are needed to address them. The work would include

- Investigation of lower-temperature operating modes
- Selection of a representative design for performance assessment purposes
- Development of additional models for thermal analysis of operational modes
- Testing to validate thermal models.

7.2 PLANS FOR SAFETY MARGIN AND DEFENSE-IN-DEPTH

Safety margin and defense-in-depth are critical to a postclosure safety case. Although not specifically called out in the proposed regulation, the risk-informed, performance-based philosophy underlying that proposed regulation is dependent upon these factors (see Appendix B).

The current postclosure safety case provides substantial margin in meeting the NRC's proposed radiological limit. Dose rates are calculated to be several orders of magnitude below the standard in the first 10,000 years and to be less than this standard for more than 80,000 years. Additional work to evaluate the dose rate beyond 100,000 years would help to assess the

robustness of postclosure performance in the first 10,000 years. Accordingly, the strategy is to conduct testing and analyses to improve key models for the estimates of longer-term mean annual dose.

The strategy focuses on two areas in this regard. First, the model for performance of the invert below the waste package will be thoroughly reviewed. The representation of the diffusion coefficient for the invert material will be evaluated to characterize uncertainties in the measurements and the extrapolation of measurements to repository conditions. If appropriate, the diffusion coefficient would be measured and the model for diffusive transport through the invert revised. Subsequent evaluation could demonstrate that the performance of the drip shield/drift invert may provide significant defense-in-depth.

Second, possible improvements in other process models that could contribute additional defense-in-depth will be investigated. Models that will be evaluated for this purpose include:

- Seepage
- Emplacement drift environments
- Effects of heat on in-package transport
- Drift shadow
- Unsaturated zone and saturated zone transport.

If the seepage can be demonstrated to be very low, if the expected unsaturated zone flux can be demonstrated to be below the threshold for seepage into the emplacement drifts, or if long-term passive ventilation can be shown to be effective in limiting moisture in the drifts, the current model could be shown to be conservative. Likewise, environments on the waste package could be much more favorable than the bounding environments currently calculated in the performance assessment modeling. The model for radionuclide migration within the waste package is conservative; reduced transport under unsaturated conditions is not considered, and the potential effects of heat on eliminating diffusive and advective transport altogether are neglected in the current model. Reduced mobility of radionuclides in the drift shadow is also neglected in the current model. Finally, the radionuclide transport models for the unsaturated and saturated zones are considered to be conservative. Additional work in these areas could demonstrate contributions of multiple barriers and additional defense-in-depth. Accordingly, feasibility of obtaining additional information useful in this regard will be evaluated. If feasible, work will be done to obtain the information.

This work will develop information to provide additional confidence that the current models either represent the system or bound effects. That is, the information will be used to argue for additional defensibility of the current models and additional margin and defense-in-depth as appropriate. Thus, most of the models used in the future TSPA analyses will not be substantially different from those considered in the current postclosure safety case, but the scientific underpinning will be strengthened, as the new information warrants. In a few cases, such as the effects of heat on radionuclide transport within waste packages, models will be developed to replace bounding assumptions.

7.3 PLANS FOR POTENTIALLY DISRUPTIVE EVENTS

The current postclosure safety case addresses potentially disruptive events. It provides a basis for excluding many of the FEPs associated with such events. In particular, nuclear criticality, effects of seismic activity, and rise of the water table are excluded as potential disruptions to the nominally expected condition (effects of credible seismic activity on cladding are taken into account in the nominal scenario however).

Potential igneous activity is not excluded and is considered explicitly in the TSPA analyses. Evaluation of igneous activity scenario currently takes account of the probability of occurrence of this event in accordance with the approach defined to estimate expected performance. However, work will be done to evaluate the conditional performance in this scenario, that is, expected performance conditioned on the event occurring. This analysis will provide additional information helpful in assessing the uncertainties associated with this scenario.

7.4 PLANS FOR NATURAL ANALOGUES

The current postclosure safety case includes information about the general nature of radionuclide transport through unsaturated volcanic rocks. In particular, this case takes into account the natural analogues discussed in Section 5.4.

Development of natural analogue information is difficult, and little new information can be obtained before submission of a license application. However, additional information can be developed over the longer term. The period after license application offers the opportunity to consider additional natural analogue work. This work could help address long-term performance of the repository system. A plan for such work will be developed and provided as part of the license application. The specific needs and value of commitment to such a long-term testing program have not yet been decided. The effort is therefore to evaluate this need and to develop a plan for work that would be useful in supporting a reasonable assurance finding on the postclosure safety case.

Candidate areas that will be evaluated include the following. The first area is continued evaluation of the Peña Blanca site. Long-term evaluation of the plume of radionuclides at this analogue site may increase confidence in the estimate of potential travel time through the unsaturated zone at Yucca Mountain. A second area is geothermal analogue data for the evaluation of the thermal-hydrologic-chemical coupled-effect component of the performance assessment model. A third area is the evaluation of long-term performance of natural nickel-containing minerals. Such information could help reinforce the conceptual model of corrosion resistance of the nickel-chromium-molybdenum alloy of the waste package outer barrier and serve to enhance confidence in the assessment of waste package performance. A fourth area is geothermal analogues to provide additional information regarding potential coupled processes.

7.5 PLANS FOR PERFORMANCE CONFIRMATION

The Performance Confirmation Plan will be updated to:

1. Provide consistency with the performance confirmation requirements of NRC's final regulation

2. Address the final principal factors of postclosure safety case supporting a license application

3. Define the relationships between the results of performance confirmation testing and a potential decision to exercise the retrieval option and the decision to close the repository.

The updated plan will focus on the final set of principal factors of the postclosure safety case. Consistent with the strategy summarized in Appendix B (see Section B.5), the performance confirmation program will be conducted in accordance with the specific regulatory requirements for monitoring, measurement, and testing with emphasis on those items identified as principal factors. This approach is consistent with the general concept of performance confirmation provided in the proposed regulation, since the principal factors are those factors that determine whether the postclosure performance objective is met.

8. CONCLUSION AND PROSPECTUS

The postclosure safety strategy has been revised to reflect enhancements in the repository system design, improved technical data and models, and changes in the regulatory framework applicable to a repository system at the Yucca Mountain site. The changes have resulted from increased understanding of what is important to postclosure performance of the repository system. That understanding has led to a focus on the following principal factors of the postclosure safety case:

- Performance of the waste package
- Performance of the drip shield/drift invert system
- Seepage into emplacement drifts
- Radionuclide concentration limits in water
- Radionuclide delay through the unsaturated zone
- Radionuclide delay through the saturated zone
- Probability of igneous intrusion
- Repository response to igneous intrusion.

This current version of the postclosure safety case is based on the best information available at the present time. It includes a comprehensive performance assessment that evaluates the factors associated with the suitability criteria of DOE's siting guidelines. This case is nearly complete. However, a small number of issues remain to be addressed to complete the case to support the site recommendation and licensing considerations. This strategy identifies the work needed to address these remaining issues:

- Complete the review of the treatment of uncertainty and conservatism in TSPA models

- Conduct additional testing and analysis of waste package degradation

- Assess coupled effects

- Refine models and conduct performance assessments to assess both above-boiling and lower-temperature options

- Evaluate the potential for the drip shield/drift invert system to provide increased defense-in-depth

- Evaluate other process models to determine potential for increased margin and defense-in-depth:

  - Seepage
  - Emplacement drift environments
  - In-package transport and transport away from the waste package
  - Drift shadow
  - Unsaturated and saturated zone transport.

- Provide conditional results for the igneous activity groundwater release scenario

- Evaluate value of studies of natural analogues of principal factors during the performance confirmation period

- Revise the Performance Confirmation Plan for a License Application

Other work might be conducted as a consequence of the results of that listed here. In addition, other work would be conducted in accordance with agreements with the NRC regarding key technical issues identified in interactions between the DOE and NRC.

Future design enhancements might suggest modifications to the current postclosure safety case. This strategy accommodates the possibility of such changes. In addition, the strategy takes into account the possibility of new information regarding process models associated with the principal factors. New information that supports or verifies that the current models are conservative would be used to increase the technical basis for the current performance assessments. New information that suggests the current models may not be conservative would be used to update those models and modify the performance assessments. However, the expectation is that, from this point forward, changes in the postclosure safety case would be minor, focusing on details of the safety case, not wholesale changes in emphasis or content.

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APPENDIX A

POSTCLOSURE SAFETY STRATEGY GLOSSARY

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APPENDIX A

POSTCLOSURE SAFETY STRATEGY GLOSSARY

Advection	The process in which solutes are transported by the motion of flowing groundwater.
Alluvium	Sedimentary material (clay, mud, sand, silt, gravel) deposited by flowing water or by wind.
Ambient	Undisturbed, natural conditions, such as ambient temperature caused by climate.
Annual Dose	The total effective dose equivalent received in a single year by the average member of the critical group as a result of release of radionuclides from the repository system.
Backfill	The material (e.g., tuff gravel, sand) placed in the excavated areas of the underground facility.
Barrier	For postclosure performance considerations, any material or structure that prevents or substantially delays movement of water and/or radionuclides.
Barrier Importance Analysis	An analysis to determine the capability of a barrier to isolate waste.
Biosphere	The ecosystem of the earth and the living organisms inhabiting it.
Biosphere Dose Conversion Factor	A multiplier used in converting a radionuclide concentration into a dose rate that a human would experience.
Breach	An opening in the waste package or drip shield caused by corrosion or mechanical stress.
Capillary Barrier	A contact in the unsaturated zone between a geologic unit containing relatively small-diameter openings and a unit containing relatively large-diameter openings. Water does not flow from the former to the latter due to capillary forces.
Cladding	The metallic outer sheath of a fuel element.
Co-disposal	Adjective that refers to waste packages that contain both U.S. Department of Energy spent nuclear fuel and defense high-level waste.
Colloid	A large molecule or small particle that has size on the order of 10^{-9} to 10^{-6} m and that is suspended in a solvent.

Commercial Spent Nuclear Fuel	Commercial nuclear fuel elements that have been removed from reactor use.
Containment	The confinement of radioactive waste within a designated boundary.
Corrosion	The process of dissolving or wearing away gradually, especially by chemical action.
Critical Group	The hypothetical group of individuals reasonably expected to receive the greatest exposure to radionuclides released from the repository system.
Criticality	The condition in which nuclear fuel sustains a chain reaction. It occurs when the number of neutrons present in one generation cycle equals the number generated in the previous cycle. The state is considered critical when a self-sustaining nuclear chain reaction is ongoing.
Defense-in-Depth	The term used to describe the property of a system of multiple barriers to mitigate unanticipated conditions, processes, and events such that the system does not rely unduly on any single barrier.
Degraded Performance Analysis	A barrier importance analysis in which barrier performance is set to an extreme value (e.g., to the 5th or 95th percentile values) to assess the sensitivity of overall system performance to the range of barrier performance.
Diffusion	The movement of a substance in a stationary material due to a gradient in concentration.
Disruptive Event	An unexpected event that results in (1) direct release of radioactivity to the surface or (2) alteration of the nominal behavior of the system.
Dose	The quantity of radiation energy absorbed in a material.
Dose Rate	See Annual Dose.
Drift	A horizontal passage in the underground facility.
Drift Invert	The element constructed in a drift to provide the floor of that drift. In an emplacement drift, this element also serves as a barrier to migration of radionuclides that might escape from breached waste packages above the drift invert.

Drip Shield	An engineered barrier that is placed above the waste package to prevent seepage water from directly contacting the waste package.
Engineered Barrier System	The waste packages and the underground facility.
External Criticality	A condition in which a critical configuration of fissile material occurs after this material is released from the waste packages. (see Criticality).
Fault (Geologic)	A fracture in rock along which movement of one side relative to the other has occurred.
Flow	The movement of a fluid such as air or water.
Flux	The rate of transfer of fluid, particles, or energy passing through a unit area per unit time.
Fractures	Breaks in rocks.
Groundwater	Water contained in pores or fractures in either the unsaturated or saturated zones below ground level.
Heterogeneity	The condition of being composed of parts or elements of different kinds resulting in spatial variation in properties.
High-Level Waste	The highly radioactive material resulting from reprocessing of spent nuclear fuel.
Host Rock	The rock unit in which the waste is emplaced.
Human Intrusion	The disturbance of a disposal system by humans that could result in release of radioactive waste.
Igneous activity	Processes or events associated with the formation and movement of molten rock.
Infiltration	The process of water entering the soil at the ground surface and the ensuing movement downward.
In Situ	In its natural position or place.
Inventory	The amount of radionuclides in radioactive waste, usually expressed in terms of mass or curies.
Invert	See Drift Invert.

Juvenile Failure	A breach in the waste package artificially set to occur early in order to provide insight into system performance. This term is distinguished from mechanistically possible early failures.
Key Technical Issues	Issues defined by the U.S. Nuclear Regulatory Commission as important for assessing the long-term safety of a potential Yucca Mountain repository.
Magma	Molten or partially molten rock material that is naturally occurring and is generated within the earth.
Matrix	Rock mass between explicitly considered fractures.
Matrix Diffusion	The process by which radionuclides move through the pores of the rock matrix under a concentration gradient.
Mean Dose Rate	Expected value of the annual dose considering the probability of the occurrence of events and the uncertainty and variability in parameter values used to estimate the annual dose.
Migration	Radionuclide movement from one location to another.
Mobile Radionuclides	Radionuclides that can move within a water system with little or no retardation.
Mobilization	The process of releasing radionuclides from the waste form.
Monte Carlo Uncertainty Analysis	An analytical method in which probability distribution for model parameters is sampled to produce a probability distribution for overall system performance.
Natural Analogues	Systems that include processes similar to those that could occur in a repository system at Yucca Mountain for which information can be obtained to evaluate long-term (e.g., millennia) or large-scale (e.g., kilometers) behavior.
Near Field	The area and conditions within the repository, including the emplacement drifts, where the natural system may be significantly impacted by the excavation of the repository and the emplacement of waste.
Neutralization	A barrier importance analysis (sensitivity study) in which the capacity of one or more barriers to limit movement of water or radionuclides is set to zero. This calculation provides insight into the importance of the barrier(s) to defense-in-depth.

Nominal Features, Events, and Processes	Those features, events, and processes expected, given the site conditions as described from current site characterization information.
Nominal Scenario	The sequence of processes and events expected to occur, that is, in the absence of potentially disruptive events.
Pathway	A potential route by which radionuclides might reach the accessible environment and pose a threat to humans.
Perched Water	A locally saturated condition that is not continuous with the water table, because there is an impervious or semipervious layer underlying the perched zone.
Percolation	The movement of water through fractures and pores of rock or soil under hydrostatic pressure and the influence of gravity. Net infiltration into the rock or soil becomes percolation, but redistribution of flow below the surface can cause the percolation flux to be different than the net infiltration directly above it.
Pluvial	In climatology, relating to former periods of abundant rains, especially in reference to glacial periods.
Postclosure	Refers to the period of time after permanent closure of the repository system.
Principal Factors	The factors that determine whether the postclosure performance objective is met.
Probabilistic	Analyses in which uncertainty in processes and events is represented through probability distributions for the parameters of those processes and events.
Process Model	A depiction or representation of a process along with any hypotheses required to describe or to explain the process.
Process Model Factor	A particular combination of features, events, and processes important to a process models.
Pyrophoricity	The spontaneous combustion of a material upon exposure to atmospheric oxygen.
Radioactive Decay	The process by which one radionuclide transforms itself to another nuclide by emission of particles or energy.
Radionuclide	An atom that is radioactive.

Radionuclide Transport	Migration of radionuclides by diffusion or advection.
Realization	The result of a simulation in which a single value is used for each parameter. The single value for a probabilistic parameter is selected by sampling its probability distribution.
Rem	The unit of equivalent radiation dose (the word "rem" is an acronym for Roentgen Equivalent Man). Not all types of radiation produce the same effect in humans. The equivalent dose takes into account the type of radiation and the absorbed dose and its effect.
Retardation	Slowing of radionuclide movement in groundwater by mechanisms that include sorption of radionuclides, diffusion into rock matrix pores and microfractures, and trapping of large colloidal molecules or particles in small pore spaces or dead ends of microfractures.
Retrieval	The act of intentionally removing radioactive waste from the underground location at which the waste had been previously emplaced for disposal.
Risk	Expected (mean) value of consequences of an undesirable process or event.
Safety Margin	The difference between expected performance and the regulatory limit for that performance.
Saturated Zone	The region below the water table where rock pores and fractures are completely filled with groundwater.
Scenario	A sequence of processes and events.
Seepage	The inflow of groundwater moving in fractures or pore spaces of permeable rock to an open space in the rock such as a drift.
Seismic	Pertaining to, characteristic of, or produced by earthquakes or earth vibrations.
Sorb	To undergo a process of sorption.
Sorption	The binding, on a microscopic scale, of one substance to another.
Spent Nuclear Fuel	Fuel that has been withdrawn from a nuclear reactor following irradiation.
Thermohydrologic	Of or pertaining to changes in groundwater movement due to the effects of changes in temperature.
Transport	See Radionuclide Transport.

Tuff	An igneous rock formed from compacted volcanic fragments from pyroclastic (explosively ejected) flows.
Unsaturated Zone	The zone of soil or rock below the ground surface and above the water table in which the pore spaces contain water, air, and other gases.
Waste Form	A generic term that refers to radioactive materials and any encapsulating or stabilizing matrix.
Waste Isolation	To set waste apart from populated areas so that releases of radionuclides to those areas meet the applicable postclosure performance objective
Water Table	The upper surface of a zone of saturation above which the majority of pore spaces and fracture openings are less than 100 percent saturated and pressure is atmospheric.
Welded	Fused by heat.
Zircaloy	A zirconium alloy used for spent nuclear fuel cladding.

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APPENDIX B

REGULATORY FRAMEWORK CONSIDERED FOR THE POSTCLOSURE SAFETY STRATEGY

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APPENDIX B

REGULATORY FRAMEWORK CONSIDERED FOR THE POSTCLOSURE SAFETY STRATEGY

B.1 INTRODUCTION

The postclosure safety strategy focuses on the postclosure performance objective specified in the regulation proposed for a potential nuclear waste repository at Yucca Mountain, Nevada. The U.S. Nuclear Regulatory Commission (NRC) has proposed its licensing criteria for this site-specific repository system in 10 CFR Part 63 (64 FR 8640). The final rule will implement applicable environmental standards that will eventually be promulgated by the U.S Environmental Protection Agency (EPA). The EPA has proposed these standards in 40 CFR Part 197 (64 FR 46976).

Also applicable to this strategy are the site suitability guidelines for evaluating geological and other aspects of the Yucca Mountain site. The U.S. Department of Energy (DOE) has proposed its revised site suitability guidelines in 10 CFR Part 963 (64 FR 67054). These guidelines include proposed criteria and a proposed methodology for determining the suitability of the Yucca Mountain site that are consistent with the NRC's proposed licensing criteria and its implementation of the applicable environmental standards.

B.2 REGULATORY STANDARD

The NRC has been moving to a more risk-informed regulatory framework, in which insights from risk assessments are used to focus on the areas of greatest risk to the public. The NRC's proposed regulation for a repository system at Yucca Mountain formalizes this approach.

The approach focuses on two concepts. First, the approach is performance-based in that it focuses on specific performance objectives. Second, the approach is risk-informed in that decision-making takes into account both the results and findings from performance assessment and other factors. This approach permits consideration of

- A broader set of concerns than might be accounted for in the performance assessment
- Rational prioritization of these concerns in terms of both performance assessment estimates and additional judgment
- A broader set of measures to address the concerns than additional development of the risk assessment methodology.

Consistent with this risk-informed, performance-based approach, the regulatory standard for postclosure performance of the repository system is reasonable assurance that the postclosure performance objective is met. This assurance is to be provided, in part, in terms of a quantitative performance assessment that considers all available knowledge, identifies the features, events, and processes (FEPs) that might affect postclosure performance of the repository system, and takes into account known uncertainties.

Assurance will also include consideration of the resiliency of the repository system with respect to postclosure performance, that is, aspects that increase confidence the postclosure performance objective is met. These include number and diversity of the barriers of the system. In general, the risk-informed, performance-based approach will rest on a firm foundation of safety margin, defense-in-depth, and other measures to enhance confidence that the regulatory standard is met (NRC 1999). The NRC has traditionally relied on independent, redundant barriers to provide assurance of postclosure safety when quantitative estimates of system performance include significant uncertainties. This reliance is reflected in the following statements of consideration that accompany NRC's proposed rule, 10 CFR Part 63 (64 FR 8640, p. 8647):

> The defense-in-depth principle has served as a cornerstone of NRC's deterministic regulatory framework for nuclear reactors, and it provides an important tool for making regulatory decisions, with regard to complex facilities, in the face of significant uncertainties. NRC also has applied the concept of defense-in-depth elsewhere in its regulations to ensure safety of licensed facilities through requirements for multiple, independent barriers, and, where possible, redundant safety systems and barriers...The NRC maintains, as it has in the past, that the application of the defense-in-depth concept to a geologic repository is appropriate and reasonable.

The remaining question is the implementation of the defense-in-depth concept. The portions of the proposed rule requiring a system of multiple barriers and an understanding of the resiliency of the geologic repository provided by those barriers are intended to ensure defense-in-depth and increase confidence that the postclosure performance objective will be achieved (64 FR 8640, p. 8650). To provide maximum flexibility in system design, the proposed rule does not place specific requirements on the degree of defense-in-depth provided by multiple barriers. Nevertheless, the NRC expectation is that DOE will rely on a combination of barriers to enhance system performance and increase confidence in demonstrating that performance. However, it is clear from the statements of consideration accompanying NRC's proposed rule that the defense-in-depth provided by these barriers should be interpreted within the risk-informed, performance-based regulatory philosophy (64 FR 8640, p. 8647):

> Traditionally, the reliance on independence and redundancy of barriers has been used to provide assurance of safety when reliable, quantitative assessments of barrier reliability are unavailable. The Commission maintains, as it has in the past, that the application of the defense-in-depth concept to a geologic repository is appropriate and reasonable. The Commission now believes, however, that its implementation, in the context of a geologic repository, should be reexamined, in light of the advancement in methods to quantitatively assess the components of a geologic repository system and with due consideration of the Commission's goal of a regulatory program and associated requirements that are risk-informed and performance-based.

Accordingly, implementation of the standard would consider quantitative analyses, compatible with the quantitative performance assessment conducted to evaluate postclosure performance of the repository system, to assess the role of the various natural barriers in providing defense-in-depth.

B.3 POSTCLOSURE PERFORMANCE OBJECTIVE

The performance objective for the repository system will be defined in NRC's regulation for that system. In absence of that final rule, DOE's standard is to provide multiple natural and engineered barriers to the movement of water and radionuclides, which can be demonstrated to limit the annual dose to an individual in Amargosa Valley such that public health and safety is protected and standards applicable for 10,000 years after permanent closure are met. In particular, the performance objective is to meet the regulatory performance objective that eventually will be established by the NRC and the applicable environmental standards implemented in its regulations. This strategy therefore focuses on the regulatory performance objective specified in NRC's proposed rule at 10 CFR 63.113 (64 FR 8640):

- The geologic repository shall include multiple barriers, consisting of both natural barriers and an engineered barrier system.

- The engineered barrier system shall be designed so that, working in combination with natural barriers, the expected annual dose to the average member of the critical group shall not exceed 25 mrem at any time during the first 10,000 years after permanent closure, as a result of radioactive materials released from the geologic repository.

- The ability of the geologic repository to limit radiological exposures to those specified shall be demonstrated through a performance assessment that excludes the effects of human intrusion.

- The ability of the geologic repository to limit radiological exposure in the event of limited human intrusion into the engineered barrier system shall be demonstrated through a separate performance assessment for a scenario defined in the proposed rule.

The NRC regulation must eventually implement the EPA standards established for the repository system, and the final criteria could well be different. The proposed EPA standards, 40 CFR Part 197 (64 FR 46976), set a limit on the annual dose to an individual of 15 mrem, somewhat lower than the limit in the NRC's proposed rule. As in the NRC proposed rule, the EPA limit applies to an individual living in Amargosa Valley and both apply for 10,000 years following permanent closure. Because the strategy focuses on a postclosure safety case that includes substantial safety margin in meeting the limit, the distinction between the proposed quantitative limits is not considered to be an important issue in Revision 4 of the strategy. The discussions of quantitative estimates of postclosure performance in Section 3 include comparisons with the standard in the NRC's currently proposed rule. However, these are intended only for perspective and do not imply an opinion about what the final standards implemented by the NRC should be.

The EPA's proposed standard also specifies a groundwater protection standard in addition to its individual protection standard. Although the performance assessments for the site recommendation considerations do address groundwater protection, Revision 4 of the repository safety strategy does not explicitly address it because the proposed NRC rule does not mention a

groundwater standard in its criteria. The strategy simply presumes that sufficient margin will be provided to meet such a standard, should it be implemented in the final NRC rule.

B.4 PERFORMANCE ASSESSMENT

The NRC's proposed rule requires that compliance with its radiological exposure limit be demonstrated through a performance assessment. The NRC defines what is meant by performance assessment at 10 CFR 63.2 (64 FR 8640):

> Performance assessment means a probabilistic analysis that:
>
> (1) Identifies the features, events, and processes that might affect the performance of the geologic repository; and
>
> (2) Examines the effects of such features, events, and processes on the performance of the geologic repository; and
>
> (3) Estimates the expected annual dose to the average member of the critical group as a result of releases from the geologic repository.

This definition is the same as that for performance assessment in DOE's proposed siting guidelines, 10 CFR Part 963, (64 FR 67054).

The specific requirements for this performance assessment are provided in the proposed rule at 10 CFR 63.114 (64 FR 8640). These requirements state that the method shall:

> (a) Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the Yucca Mountain site, and the surrounding region to the extent necessary, and information on the design of the engineered barrier system, used to define parameters and conceptual models used in the assessment.
>
> (b) Account for uncertainties and variabilities in parameter values and provide the technical basis for parameter ranges, probability distributions, or bounding values used in the performance assessment.
>
> (c) Consider alternative conceptual models of features and processes that are consistent with available data and current scientific understanding, and evaluate the effects that alternative conceptual models have on the performance of the geologic repository.
>
> (d) Consider only events that have at least one chance in 10,000 of occurring over 10,000 years.
>
> (e) Provide the technical basis for either inclusion or exclusion of specific features, events, and processes of the geologic setting in the performance assessment. Specific features, events, and processes of the geologic setting

must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

(f) Provide the technical basis for either inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers in the performance assessment, including those processes that would adversely affect the performance of natural barriers. Degradation, deterioration, or alteration processes of engineered barriers must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

(g) Provide the technical basis for models used in the performance assessment such as comparisons made with outputs of detailed process-level models and/or empirical observations (e.g., laboratory testing, field investigations, and natural analogs).

(h) Identify those design features of the engineered barrier system and natural features of the geologic setting that are considered barriers important to waste isolation.

(i) Describe the capability of barriers, identified as important to waste isolation, to isolate waste, taking into account uncertainties in characterizing and modeling the barriers.

(j) Provide the technical basis for the description of the capability of the barriers, identified as important to waste isolation, to isolate waste.

Two of these items require special consideration in the strategy. The first of these is the event-screening criterion of Item (d). This item presents two important issues. The first is the definition of the probability of the event. In principle, the probability may be uncertain, so the question arises as to what aspect of the probability distribution reflecting this uncertain should be used in the screening. The most obvious aspect is the mean value of this probability distribution. Another possibility is an upper bound (e.g., the 95th percentile of the probability distribution) to ensure that this uncertainty is addressed. A difficulty in this approach is arbitrariness in the specification of the bound. The mean, or average value, is well defined. Further, the criterion already has the vestige of an average value in that it considers the probability over 10,000 years rather than an annual frequency. Accordingly, the criterion is interpreted here to be expressed in terms of the mean probability of occurrence. The second issue is the approach to defining what is meant by an event. In principle, an event with a probability of occurrence exceeding this criterion can be subdivided into a set of events, each of which has a probability of occurrence meeting this criterion. Care is therefore needed to ensure the definition of the event that is considered in a screening evaluation is defined broadly enough that events important to the estimate of mean annual dose are not inadvertently screened out. In particular, events that initiate a disruptive event scenario will be carefully considered.

Special considerations are also needed for Item (i), the requirement to describe the capability of barriers to isolate waste. First, this description is used to provide insight into the performance

assessment and to increase the transparency of the quantitative analysis. Second, this description is used to assess the contribution of the barriers to the degree of defense-in-depth provided by the system. Accordingly, the approach to analysis of the importance of barriers must be sufficiently comprehensive to accommodate both of these considerations.

B.5 PERFORMANCE CONFIRMATION

A key element of the postclosure safety case is performance confirmation. The principal focus of this element is to help provide reasonable assurance the postclosure performance objective of the regulation is met. Performance confirmation is defined in the proposed rule (see 10 CFR 63.2 [64 FR 8640]) as the program of tests, experiments, and analyses conducted to evaluate the accuracy and adequacy of the information used to determine with reasonable assurance that the performance objective at proposed 10 CFR 63.113(b) is met.

The proposed rule gives the general concept of the performance confirmation program at 10 CFR 63.102(m) (64 FR 8640):

> Performance confirmation. A performance confirmation program will be conducted to verify the assumptions, data, and analyses that support the performance assessment, and any findings, based thereon, that permitted construction of the repository. Key geologic, hydrologic, geomechanical, and other physical parameters will be monitored throughout site characterization, construction, emplacement, and operation to detect any significant changes in the conditions assumed in the performance assessment that may affect compliance with the performance objective at § 63.113(b).

Accordingly, the performance confirmation program will focus on the principal factors of the postclosure safety case, those factors that determine whether the postclosure performance objective is met.

In addition to the general concept of performance confirmation, specific requirements on the performance confirmation program are proposed in Subpart F of 10 CFR Part 63 (64 FR 8640). The general requirements are proposed in 10 CFR 63.131 of Subpart F. These include requirements to confirm subsurface conditions and changes in those conditions are within limits assumed in the licensing review. The proposed rule specifies that the performance confirmation program is to begin during site characterization and continue until permanent closure. Consistent with the general concept specified at 10 CFR 63.102(m), this requirement is interpreted in the current strategy for performance confirmation to mean that the in-situ monitoring, laboratory, and field testing conducted within this program would focus on those conditions determined in the performance assessment analyses to be important to postclosure performance.

Requirements on confirmation of geotechnical and design parameters are proposed at 10 CFR 63.132 of Subpart F (64 FR 8640). These proposed requirements for a program of testing to ensure geotechnical and design parameters are confirmed, including relevant measurements of rock deformation, rock stress, and other rock properties and conditions, and thermomechanical response of the repository resulting from excavation and heat generated by the emplaced waste. In the spirit of the performance confirmation concept specified at 10 CFR 63.102(m), this

proposed requirement is interpreted within the strategy for the performance confirmation program to mean that the measurements should focus on those geotechnical and design parameters determined to be important to postclosure performance.

Additional requirements on design testing are proposed at 10 CFR 63.133 of Subpart F (64 FR 8640). The principal focus of these requirements is the testing of features such as borehole and shaft seals, backfill, and the thermal interaction effects of the waste packages, backfill, rock, and groundwater. The current design does not include backfill. Furthermore, neither borehole and shaft seals nor thermal interactions of the waste packages, rock, and groundwater are considered to be important to postclosure performance. The strategy for the performance confirmation program is that the program will be conducted in accordance with all specific requirements of the final rule. However, in view of the fact that these particular features are not important to meeting the postclosure performance objective and in view of the fact that this requirement is a proposal at this time, plans for testing them are not provided in the current strategy for performance confirmation.

Requirements on monitoring and testing of waste packages are found at 10 CFR 63.134 of Subpart F (64 FR 8640). Waste package performance is a principal factor of the postclosure safety case and current performance confirmation strategy is that this monitoring and testing would be addressed in the performance confirmation program.

B.6 SITE SUITABILITY DETERMINATION

DOE's proposed siting guidelines require an evaluation of postclosure suitability by means of performance assessment. In particular, these proposed guidelines indicate at 10 CFR 963.16 (64 FR 67054) that suitability of the site could be determined through an assessment of whether the repository system is likely to comply with the applicable radiation protection standard (see Section B.3). The evaluation is to use the characteristics of the repository system identified in the criteria at 10 CFR 963.17. These criteria include

(1) Site characteristics, including (i) geologic, (ii) hydrologic, (iii) geophysical, and (iv) geochemical properties of the site

(2) Unsaturated zone flow characteristics, including (i) climate, (ii) infiltration, (iii) unsaturated zone flux, and (iv) seepage

(3) Near-field environment characteristics, including (i) thermal hydrology and (ii) near-field geochemical environment

(4) Engineered barrier system degradation characteristics, including (i) engineered barrier system component performance and (ii) waste package degradation

(5) Waste form degradation characteristics, including (i) cladding degradation and (ii) waste form dissolution

(6) Engineered barrier system degradation, flow and transport characteristics, including (i) colloid formation and stability and (ii) engineered barrier transport

(7) Unsaturated zone flow and transport characteristics, including (i) unsaturated zone transport and (ii) thermal hydrology

(8) Saturated zone flow and transport characteristics, including (i) saturated zone transport and (ii) dilution

(9) Biosphere characteristics, including (i) reference biosphere and receptor and (ii) biosphere transport and uptake.

The process model factors encompass all of the FEPs relevant to postclosure performance of a repository system at Yucca Mountain. The correlation between the process model factors and the criteria of 10 CFR Part 963 (64 FR 67054) is indicated in Table B-1. Because these process model factors include all FEPs that are credible for a repository system at the Yucca Mountain site, analyses of the set of factors in this table effectively addresses all the associated criteria of 10 CFR 963.17. Consequently, the analyses of the process model factors address all of these criteria.

These criteria dictate a postclosure safety case different from one that focuses solely on the requirements of NRC's proposed rule. While the requirements of DOE's site suitability guidelines call for a judgment of the likelihood of meeting the performance objective of the NRC rule, this judgment must explicitly consider each of the system characteristics identified in 10 CFR 963.17 and not merely the characteristics determined to be most important to postclosure performance. While the difference is subtle, the postclosure safety case needed to support a site suitability evaluation must consider all characteristics without regard to their importance. This postclosure safety case, therefore, may provide a broader set of considerations than the case that would be prepared for a possible license application.

Table B-1. Correlation between Process Model Factors and Criteria of 10 CFR Part 963

Process Model Factors	Related Criteria of 10 CFR Part 963	
	Sections	Criteria
Climate	10 CFR 963.17(1) and (2)(i)	Site characteristics, Climate
Net infiltration	10 CFR 963.17(1) and (2)(ii)	Site characteristics, Infiltration
UZ flow	10 CFR 963.17(1) and (2)(iii)	Site characteristics, UZ flux
Repository-induced effects on UZ flow	10 CFR 963.17(1) and (3)(i)	Site characteristics, UZ thermal hydrology
Seepage into emplacement drifts	10 CFR 963.17(1) and (2)(iv)	Site characteristics, Seepage
Repository-induced effects on seepage	10 CFR 963.17(1) and (3)(i)	Site characteristics, Thermal hydrology
In-drift physical and chemical environments	10 CFR 963.17(1) and (3)(ii)	Site characteristics, Near field geochemical environment
In-drift moisture distribution	10 CFR 963.17(1) and (3)(i)	Site characteristics, Thermal hydrology
Waste package degradation and performance	10 CFR 963.17(4)(ii)	Waste package degradation
Drip shield degradation and performance	10 CFR 963.17(4)(i)	Engineered barrier system component performance
Cladding degradation and performance	10 CFR 963.17(5)(i)	Cladding degradation
CSNF degradation and performance	10 CFR 963.17(5)(ii)	Waste form dissolution
HLW degradation and performance	10 CFR 963.17(5)(ii)	Waste form dissolution
DSNF degradation and performance	10 CFR 963.17(5)(ii)	Waste form dissolution
Dissolved radionuclide concentrations	10 CFR 963.17(5)(ii)	Waste form dissolution
Colloid associated radionuclide concentrations	10 CFR 963.17(6)(i)	Colloid formation and stability
EBS (invert) degradation and performance	10 CFR 963.17(6)(ii)	EBS component performance and transport
UZ radionuclide transport	10 CFR 963.17(1) and (7)(i)	Site characteristics, UZ transport
Repository-induced effects on UZ radionuclide transport	10 CFR 963.17(1) and (7)(ii)	Site characteristics, UZ thermal hydrology
SZ flow	10 CFR 963.17(1) and (8)(ii)	Site characteristics, SZ dilution
SZ radionuclide transport	10 CFR 963.17(1) and (8)(i)	Site characteristics, SZ transport
Biosphere dose conversion factors	10 CFR 963.17(9)	Biosphere transport and uptake

Source: 64 FR 67054. EBS = engineered barrier system.

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APPENDIX C

METHODOLOGY FOR ANALYSES USED IN THE POSTCLOSURE SAFETY CASE

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APPENDIX C

METHODOLOGY FOR ANALYSES USED IN THE POSTCLOSURE SAFETY CASE

This appendix describes the methodology used for performance assessments and other analyses in the postclosure safety case. The methods for performance assessment exclude the effects of human intrusion, barrier importance analyses, and separate performance assessment of inadvertent human intrusion. These methods are consistent with the regulations that are summarized in Appendix B. A general reference for this information can be found in Section 3 of *Total System Performance Assessment—Site Recommendation* (CRWMS M&O 2000h).

C.1 METHODOLOGY FOR PERFORMANCE ASSESSMENT

This section describes the method being developed to estimate expected postclosure performance. Analyses have been conducted for the repository system since 1991. These analyses have steadily improved because of improvement in site and design information and increased sophistication of the model abstractions and methodology.

The method is probabilistic, applicable to all process and component models and their linkage for the estimate of postclosure repository system performance. The postclosure performance measure computed is the annual dose to an average member of a critical group. The method permits an estimate of the mean annual dose rate the variance around the mean arising from uncertainty in the component models and their parameters.

The performance assessment method takes full account of current understanding of phenomena relevant to the postclosure performance measure. In particular, the method considers the information specified in the proposed rule 10 CFR Part 63 (see Appendix B). At the outset, the consideration includes those aspects implied at 10 CFR 63.114 (a) through (d):

- Data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the Yucca Mountain site, and the surrounding region, and information on the design of the engineered barrier system, used to define parameters and conceptual models used in the assessment

- Uncertainty and variability in parameter values

- Alternative conceptual models of features and processes that are consistent with available data and current scientific understanding

- Events that have at least one chance in 10,000 of occurring over 10,000 years.

These considerations include an evaluation of all features, events, and processes potentially important to the postclosure performance measure. In particular, credible processes or events (those with a probability of occurring in the 10,000-year period following permanent closure of greater than one chance in 10,000) that could degrade or alter engineered or natural barriers important to performance are explicitly taken into account in the performance assessment. They also ensure that potentially favorable processes are not given undue credit in the performance assessment.

Evaluation of the features, events, and processes of potential importance to the estimate of annual dose to an individual in Amargosa Valley has led to three types of scenarios. The first of these is the nominal scenario, which includes the sequence of processes and events expected to occur at Yucca Mountain but excludes igneous activity and human intrusion. The second scenario includes igneous activity in terms of intrusion of magma into the repository emplacement drifts in the next 10,000 years. This event has very low probability and is not expected to occur; nevertheless, volcanism is sufficiently credible to warrant consideration. The third scenario includes the possibility of inadvertent human intrusion.

Models are developed for each scenario, to represent system components potentially important in each case. These models are developed from detailed process models for the major subsystems. The component models are implemented in computer codes.

Parameters for the component models are developed from the description of the process models and are based on measurements conducted to develop those models. Uncertainty in these models is accounted for in terms of probability distributions for the parameters. In some cases, the probability distributions are based on the statistical distribution of measured values. In other cases, the distribution is estimated based upon general considerations and inferences from the models. Each distribution describes a range within which the true value is believed to fall, with a mean or median value corresponding to the best estimate of the true value. Not all parameters in the performance assessment require uncertainty distributions. Single values are used to represent properties for which uncertainty is known to have little or no effect on overall performance. In addition, where appropriate, single values may be used to bound uncertainty in selected parameters.

The computer codes for the component models are linked to allow simulation of the overall system behavior. Uncertainty associated with the model parameters is included in the simulation by conducting multiple calculations for each scenario, using values sampled from ranges of possible values. Each individual calculation uses a different set of sampled input values. The result of each calculation represents a different possible "realization" of future postclosure system performance consistent with the uncertainty in the input parameters. Each realization represents the calculation for each scenario weighted by the probability that the scenario will occur. The expected behavior of the system is indicated by the mean of the set of calculations, and the range of calculated behaviors indicates the uncertainty associated with that mean. The performance assessment is, therefore, a probabilistic analysis consistent with the requirements of the U.S. Nuclear Regulatory Commission's (NRC's) proposed regulation (see Appendix B), in that it takes into account both the estimates of the probability of different scenarios and the uncertainty associated with input parameters.

Results of performance assessments can be analyzed at both the system and subsystem levels to identify the models and parameters with the greatest effect on system behavior. These analyses include sensitivity and uncertainty analyses to evaluate performance over the entire range for the parameters. These include parametric studies to examine the importance of model assumptions, simplifications, and limitations.

C.2 METHOD FOR BARRIER IMPORTANCE ANALYSES

NRC's proposed rule requires identification of key system barriers and a description of their contribution to postclosure performance assessment (see Appendix B). This contribution is evaluated by conducting analyses in which the performance of one or more barriers is reduced to see the effect of this reduction on overall postclosure system performance. The difference between the results in which the barrier performs as expected and in which its performance is reduced provides an indication of the contribution of that barrier to overall performance.

The barrier importance analyses serve several purposes. In addition to providing analyses of the contribution of the principal barriers, the method effectively unravels the performance assessment and enables insight into importance of individual models and barrier representations in a particular calculation. The result is a better understanding of the impact of model uncertainties and assumptions and simplifications in the performance assessment. In addition, the analyses indicate the degree to which various barriers compensate for one another in terms of independence, redundancy, and effectiveness and therefore provides an overall indication of the degree of defense-in-depth provided by the system (see Section C-4). The analyses also provide input to the determination of the principal factors of the postclosure safety case.

The method considers two approaches to treatment of reduced barrier performance: "degraded" performance and "neutralized" performance. In the first approach, barrier performance is set to an extreme value within its range of possible performance. For example, parameters of the model for the barrier's ability to prevent radionuclide movement (e.g., transport retardation factors) are set to their 5th or 95th percentile values. This approach is used to assess the sensitivity of overall postclosure system performance to the range of uncertainty in barrier performance. It provides insight into the importance of the uncertainties that are quantified in the base case calculation.

The degraded barrier approach, however, does not provide a complete assessment of the contribution of the barrier. For example, if a barrier's performance is certain, the analyses would show no effect whatsoever. Therefore, in addition to degraded barrier analysis, calculations are made completely neutralizing the performance of the barrier. In this approach, all capacity of the barrier to limit movement of water or radionuclides is set to zero. The resulting calculation has no bearing on actual performance of the system because complete neutralization of a barrier generally is impossible. The approach is used simply to gain insight into the importance of the barrier. For example, setting the lifetime of an engineered barrier to zero provides insight into the importance the actual lifetime of this barrier might have and the importance of uncertainties in the information leading to the determination of that lifetime.

Neutralization of a single barrier may not provide insight into the actual contribution of a barrier. If barriers are truly redundant, neutralization of one of them could result in little change in system performance. In order to gain full insight, therefore, the analyses may need to consider neutralization of combinations of barriers. Neutralized barrier analyses are therefore needed to address defense-in-depth issues.

Finally, neutralization analyses are needed to identify the factors critical to the determination of postclosure performance. If neutralization of a factor produces a significant change in the estimate of annual dose, the factor can be identified as a principal factor.

In the neutralization analysis, the specific function or factor at issue is neutralized or set to zero. The approach is illustrated schematically in Figure C-1. The upper curve shows an estimate of system performance in which no barriers contribute (such as in the upper curve of Figure 3-2). The next curve shows the effect calculated when some of the barriers are assumed to function (e.g., limit movement of water or radionuclides) and are included in the analysis. System performance is improved (i.e., annual dose is reduced) in this case. The lowest curve shows the estimate when all barriers are included in the analysis. This calculation represents expected performance of the total system of multiple barriers. The second curve then represents a neutralization analysis, in which the barriers not included in the calculation are neutralized. The upper curve is schematic of another neutralization analysis, one in which all of the barriers are neutralized.

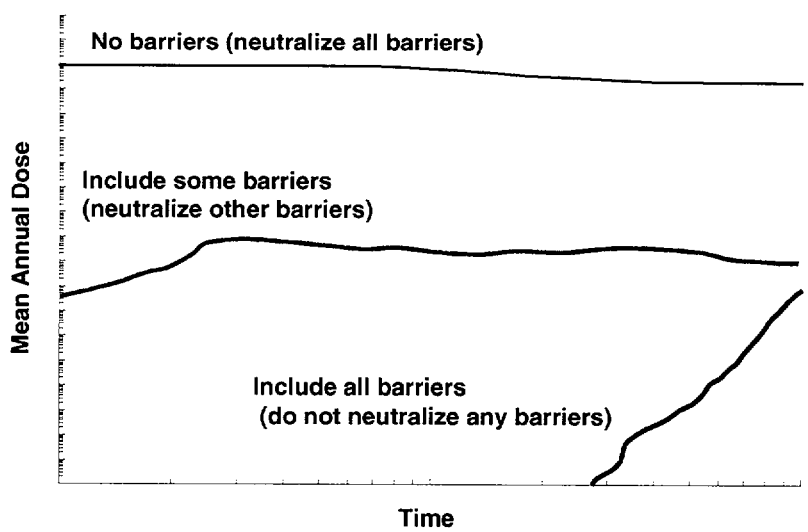

Figure C-1. Illustration of Neutralization Analysis

These neutralization analyses do not represent performance of the system under extreme conditions. They are used simply to determine contributions of various factors to an estimate of expected performance of a system of barriers.

These analyses can also be used to assess the degree of defense-in-depth provided by the system. Defense-in-depth means that barriers are redundant, that is, perform the same function, such that failure of any one of them does not cause failure of the system. In the case of perfect redundancy, neutralization of a single barrier would show no significant change to system

performance. In this case, complete assessment of the system requires neutralization of combinations of barriers or factors, in addition to the analyses of individual barriers or factors.

A schematic illustration of a defense-in-depth analysis is shown in Figure C-2. The illustration shows the estimate of expected performance for a complete system of barriers. It also shows the neutralization analysis for each of two barriers. The figure also shows simultaneous neutralization of both barriers. The combination of all three analyses is used to assess whether or not barriers are redundant.

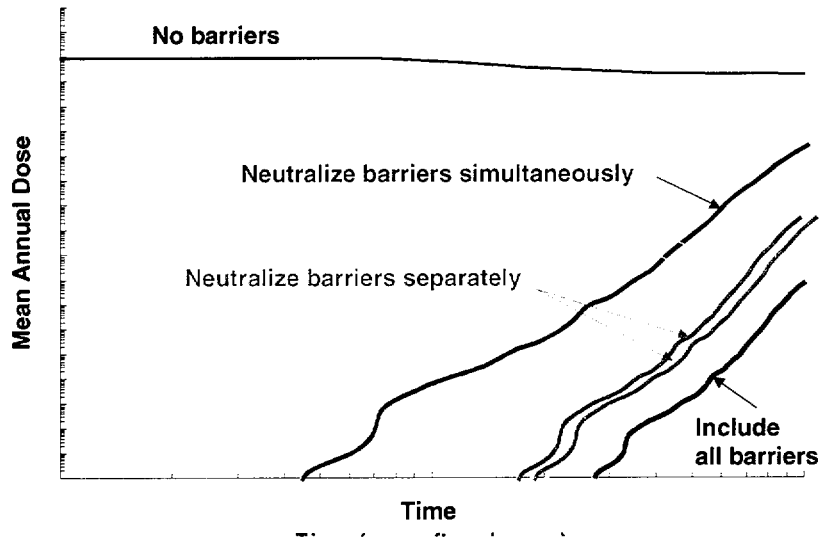

Figure C-2. Illustration of Defense-in-Depth Analyses

C.3 METHOD FOR ASSESSMENT OF HUMAN INTRUSION

Future human activity that might interfere with the repository cannot be precluded because such activity thousands of years into the future cannot be predicted. It is not practical to examine every possible way human activity could interfere with a repository. The NRC's proposed rule specifies a particular scenario for evaluation to indicate the risk associated with such activity (see Appendix B). The approach taken to address this human intrusion scenario assumes the following:

- The event would occur 100 years after permanent closure.
- The event would involve a nearly vertical borehole through a single waste package (and the associated drip shield) to the water table.
- Borehole diameter and composition of drilling fluids are consistent with resource exploration

- The borehole would not be sealed but would include cuttings and other material (properties assumed to be similar to a fault pathway)

- Groundwater is the only pathway considered

The intersection of the borehole with the waste package is represented as two (top and bottom) waste package breaches. Transport of the mobilized radionuclides is then represented by advection and diffusion from the waste package to the water table through the material in the borehole. Transport in the water table is represented by the advection in the saturated zone (SZ) flow system.

Most models for this assessment are the same as those used for the performance assessment that excludes effects of human intrusion. These include the models for mobilization of radionuclides in the intersected waste package and transport of the radionuclides in the SZ flow system. The flow rate in the borehole is greater than would be estimated using the average infiltration rate to account for focusing flow into the borehole. The analysis includes more radionuclides than considered for the performance assessment without human intrusion, because the borehole penetration of the waste package and the unsaturated zone permits release of radionuclides that would decay before these are normally breached. Another difference is that the models for transport of radionuclides away from the waste package take into account the borehole configuration and properties. All of these models are linked just as for the performance assessment that excludes effects of human intrusion. The performance measure is calculated in a similar fashion. The results are not integrated with those for the nominal and igneous activity scenarios but are considered separately.

C.4 METHOD FOR SAFETY MARGIN ANALYSIS

Figure C-3 illustrates the approach to determine the degree of safety margin provided by the system. The approach is to compare the mean annual dose for the system with the radiological exposure limit in two ways. First, the peak of the mean annual dose in the first 10,000 years is compared with the numerical limit. The difference indicates the margin in this sense. If this difference is small, then uncertainties in estimate of the mean annual dose could result in a future estimate exceeding the limit.

The time that the calculated mean annual dose equals the radiological exposure limit is also evaluated. The difference between this time and 10,000 years indicates margin in the second sense. If this difference is small, then uncertainties in the time radionuclides reach the receptor in Amargosa Valley (e.g., because of uncertainties in waste package lifetime or uncertainties in travel time through the unsaturated zone or the saturated zone) could result in the mean annual dose in a future calculation exceeding the limit within 10,000 years.

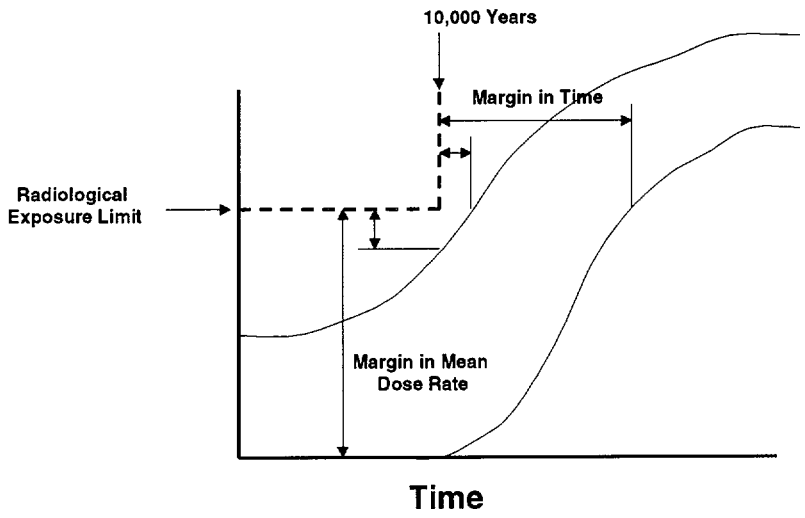

Figure C-3. Schematic of Safety Margin in Mean Dose Rate and Time

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APPENDIX D

GENERAL APPROACH TO ENHANCEMENT OF CONFIDENCE IN POSTCLOSURE SAFETY

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APPENDIX D
GENERAL APPROACH TO ENHANCEMENT OF CONFIDENCE IN POSTCLOSURE SAFETY

D.1 NUCLEAR ENERGY AGENCY PAPER ON CONFIDENCE IN THE LONG-TERM SAFETY OF DEEP GEOLOGICAL REPOSITORIES

The treatment of uncertainty is critical to the postclosure safety case. The treatment of uncertainty and other potential vulnerabilities considered in Revision 4 of the repository safety strategy is consistent with the approach outlined in Chapter 4 of the Nuclear Energy Agency paper, *Confidence in the Long-term Safety of Deep Geological Repositories—Its Development and Communication* (Nuclear Energy Agency 1999). This paper gives the general considerations regarding uncertainty:

> Uncertainty is the result of limited knowledge. In evaluating the evolution of the disposal system in performance assessment, uncertainties in the available scientific understanding, models, and data are inevitable, due to system complexity and the long-time scales involved. The result of these uncertainties is a corresponding uncertainty in the evaluated performance of the system. If the degree of uncertainty in the evaluated performance is such that the confidence in the safety that it indicates is judged to be unacceptable, then a number of measures can be employed to enhance confidence. The aim of these measures is to generate an assessment in which, while acknowledging the presence of uncertainty, there is sufficient confidence, or reasonable assurance, in the safety that it indicates to support a positive decision (by the implementer) to compile and present a safety case, and subsequently, a positive decision (by the regulator) to proceed to the next stage of repository development.

These measures fall into two groups:

- Those that aim to increase the robustness of the system concept
- Those that aim to increase the quality of the performance assessment.

Robustness in the system concept facilitates the perception of intrinsic safety and the evaluation and communication of safety. Measures that aim to increase the robustness of the system concept generally reduce sensitivity to the presence of any unresolved issues and uncertainties. Particular measures identified in the paper include the following:

- The adoption of safety provisions that give rise to a robust concept, in which uncertainties are avoided or safety can be demonstrated in their presence. This includes application of safety margin and defense-in-depth.

- The adoption of a flexible strategy to design development and improvement in order to ensure efficient use of the safety potential of the host rock (e.g., "design-as-you go")

Quality of the performance assessment includes quality of the assessment capability and the reliability of its application in performance assessment. This quality depends upon the degree to

which the assessment takes into account current understanding of phenomena relevant to long-term safety and the uncertainty in these phenomena. Measures identified in the paper include the following:

- Increase confidence in completeness of the FEPs in the scenarios included in the assessment, in particular, those FEPs that could lead to increased consequences.
- Quantify key uncertainties in model parameters, recognizing that quantification of some uncertainties may not be practicable, such as those associated with human intrusion and the treatment of the biosphere.
- Conduct sensitivity and uncertainty analyses to enhance transparency and provide insight into the importance of the uncertainties that have been quantified.

D.2 U.S. DEPARTMENT OF ENERGY APPROACH TO THE EVALUATION AND MANAGEMENT OF UNCERTAINTY

DOE's general approach to the treatment of uncertainty is shown schematically in Figure D-1.

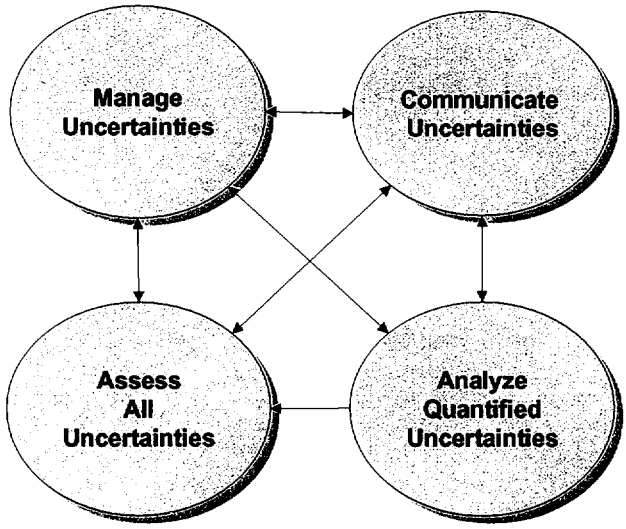

Figure D-1. General steps to address uncertainty and other potential vulnerabilities in the postclosure safety case

The Nuclear Energy Agency paper recommends an iterative program of reassessment to take into account new information, changes in repository design, and increased performance assessment capability. DOE's general approach to uncertainty is consistent with this recommendation.

The structure shown in Figure D-1 illustrates the specific elements of this process. These elements address both quantified and unquantified uncertainties. The process is iterative and has proceeded since the *Site Characterization Plan, Yucca Mountain Site, Nevada Research and Development Area, Nevada* (DOE 1988) and its issue resolution strategy. The process begins by assessing uncertainties and proceeding to other steps. The arrows indicate information feeding from each element of this strategy to others. Each of these elements and the information feeds are explained in the following sections.

D.2.1 ANALYZE QUANTIFIED UNCERTAINTIES

This element includes analyses to provide input to general assessment of uncertainties. Through iteration, these analyses are modified as a result of changes in the strategy and feedback. The uncertainties considered in this element are those that can be quantified and addressed within the context of the total system performance assessment (TSPA) methodology. These include uncertainties arising from the following:

- Process model complexity (e.g., conceptual and mathematical model uncertainty)
- Variability and parameter uncertainty
- Low-probability, high-consequence mechanisms (e.g., potentially disruptive events).

These uncertainties are explicitly addressed within the TSPA analyses. For example, model and parameter uncertainty is captured in terms of probability distributions for the parameters of the models in the TSPA analyses. These probability distributions reflect the variation in measured values and beliefs about the degree of uncertainty in the representations themselves. Potentially disruptive processes and events are addressed through scenarios that explicitly incorporate these processes and events. The scenario probability is taken into account so that the risks from these low-probability occurrences are given proper weight. Sensitivity and importance analyses are conducted to identify areas where the uncertainties are most important and to provide an estimate of the degree of this importance within the context of the uncertainty estimates. The uncertainty itself is analyzed in terms of variances and contribution to the overall variance of the performance estimates.

D.2.2 ASSESS ALL UNCERTAINTIES

This element addresses all uncertainties, including those that cannot be quantified within the context of the TSPA methodology. The objective of this step is to provide input to management consideration of how uncertainty is to be managed. The element considers input from the analyses of quantified uncertainty. The assessment in this element considers the limits to the TSPA analyses, including the limits in the models, probability estimates, and scenario representations. The assessment also includes evaluation of known, but unincorporated, uncertainty (e.g., centimeter scale heterogeneity and effect of nonlinear friction forces in the flow and transport). It also includes an assessment of the potential for unknown failure modes or other aspects not explicitly addressed in the performance assessment models and the potential for new findings about site characteristics and materials behavior.

D.2.3 MANAGE UNCERTAINTIES

This element describes the measures taken to address uncertainty. The strategy in this case focuses on the principal factors of the system, identifies the critical uncertainties, and the approaches to reducing or mitigating them. The various options for addressing the critical uncertainties are evaluated in terms of such factors as:

- Magnitude and importance of uncertainties
- Introduction of new problems/uncertainties
- Feasibility of the options
- Effectiveness in addressing the uncertainties
- Demonstrability
- Cost for each option.

Options for reducing or mitigating uncertainty include the following:

- Additional information
- Credit/conservatism in the analyses
- Design enhancement (e.g., to increase safety margin and defense-in-depth).
- Other measures that build confidence including the following:
  - Explicit discussion of key disruptive events
  - Insights from natural analogues
  - Commitment to specific future confirmation tests.

D.2.4 COMMUNICATE UNCERTAINTIES

Finally, the process includes communication of the approach to addressing uncertainty to decision-makers and interested parties and developing feedback on this approach. The element includes communication of the sources of uncertainty, the magnitudes of the associated uncertainty, and their potential impact on postclosure performance. It includes communication of the measures and their expected effectiveness. It includes formal documentation and informal communication.