Knowledge Management Roadmap of the Regulatory Development and Safety Review for Geologic Disposal of High-Level Waste

Contents

1.0 Introduction and Purpose

This knowledge management Roadmap was developed by the U.S. Nuclear Regulatory Commission (NRC) technical staff to preserve relevant knowledge gained over many years of preparation and licensing review of the proposed geologic repository at Yucca Mountain, as described and approved by the Commission in COMSECY-20-0013 and the accompanying SRM. In particular, The Roadmap provides the references and their context for NRC's high-level waste (HLW) disposal program development path beginning with the development of the regulatory requirements, preparations to review the license application (pre-licensing period) and through to the completion of the staff's Safety Evaluation Report (SER) for a potential repository at Yucca Mountain. This report outlines the technical documentation that includes the regulatory framework and the technical review. Given the length of time since any significant activities have been conducted with respect to licensing activity for Yucca Mountain, this document was developed to capture the information obtained over the Part 63 rulemakings and staff evaluation of the Yucca Mountain License application. This knowledge management document is limited to the purposes outlined in the COMSECY; technical information has also been captured for specific topics related to Yucca Mountain (information available at https://www.nrc.gov/waste/hlw-disposal/historical-information#km).

Consideration of regulatory approaches for evaluating and authorizing geologic disposal for high-level radioactive waste (HLW) at the U.S. Nuclear Regulatory Commission (NRC) began in the 1970s. [In this context, the term HLW encompasses commercial spent nuclear fuel (SNF), various U.S. Department of Energy (DOE) SNFs, and DOE high-level glass waste.] Since that time, the NRC:

- completed its generic regulations for a HLW repository (10 CFR Part 60 – finalized in the early 1980s),
- promulgated specific regulations for a HLW repository at Yucca Mountain, Nevada (10 CFR Part 63 – finalized in 2001 and later revised to include regulations for the period after 10,000 years up to 1-million years that were finalized in 2009),
- accepted for review the U.S. Department of Energy's (DOE) license application for a potential repository at Yucca Mountain in 2008,
- completed the NRC staff's Technical Evaluation Report for a potential repository at Yucca Mountain (published in 2011), and
- prepared the NRC staff's Safety Evaluation Report for a potential repository at Yucca Mountain in 2015.

This document is intended to serve as a 'roadmap' to help staff understand the HLW disposal program with respect to the (1) overall approach for the regulations and the safety review of the proposed repository at Yucca Mountain, and (2) resulting technical insights and perspectives resulting from these development and review activities. Documentation of this information is considered useful for informing any future regulatory activities for the geologic disposal of HLW. For example, should the Yucca Mountain proceeding move forward this document would help inform staff new to the HLW disposal program and serve as a 'refresher' to other staff with previous experience with the HLW disposal program. Should the National Program move forward with either a new site or disposal concept, this document is intended to provide an understanding for the regulations for geologic disposal and the important safety aspects of geologic disposal that would inform any future regulatory activities such as revisions to NRC's generic regulations for HLW disposal at 10 CFR Part 60 and preparations for the safety review (e.g., technical areas and allocation of resources prior to the review). Although other efforts have been conducted to capture technical information regarding specific areas of the safety evaluation (e.g., infiltration models, corrosion tests) this document provides perspectives and insights across the entire time period of regulatory development and the safety review as documented in the staff's SER for Yucca Mountain.

This report is not intended to re-interpret or revise the SER – and to the extent practicable this report utilizes direct quotes from the SER (identified in most cases by indented and italicized text for easy identification). The body of this report is considered a historical summary of the key aspects of the regulatory development and safety review. The report does contain two appendices that provide further details on the regulatory development (Appendix A) and postclosure safety (Appendix B).

2.0 Yucca Mountain Timeline

Programs for the geologic repository for high-level radioactive waste (HLW) disposal were developed over decades during which there have been a number of key decisions/milestones as the National program progressed. This development over the years has included legislative action, environmental standards by the U.S. Environmental Protection Agency (EPA), regulations for a HLW repository by the NRC, and the license application for the repository by the U.S. Department of Energy (DOE). Some of these key decisions and milestones are summarized below:

<u>1980 to 1989</u>

- The Nuclear Waste Policy Act of 1982 (NWPA) provided for the development of repositories for the disposal of HLW (e.g., stated HLW disposal is a Federal responsibility, established schedule for siting, construction, and operation of repositories; set limit for first repository of 70,000 metric tons heavy metal (MTHM); assigned responsibilities to DOE, EPA, NRC, the President, and to Congress; and established the Nuclear Waste Fund). In particular, the NWPA specified that NRC criteria for spent fuel and HLW disposal must: provide for a system of multiple barriers, include restrictions on retrievability, and not be inconsistent with the generally applicable EPA standards. Additionally, the NRC has obligations under the NWPA to consult with DOE prior to licensing (e.g., comment on site characterization plan and progress of characterization activities, make preliminary comments on sufficiency of site characterization and DOE's waste form proposal prior to site recommendation).

- The Nuclear Waste Policy Amendments Act of 1987 (NWPAA) specified the characterization of a single site at Yucca Mountain, Nevada. If Yucca Mountain cannot be licensed, DOE must seek Congressional direction.

<u>1990 to 1999</u>

- The Energy Policy Act of 1992 directed EPA to develop site-specific, health-based standards for evaluating a potential Yucca Mountain repository (including contracting with the National Academy of Sciences for recommendations for Yucca Mountain Standards). NRC was directed to revise its technical criteria to be consistent with EPA's site-specific standards for Yucca Mountain.

- The National Academy of Sciences published its report on the Technical Bases for Yucca Mountain Standards in 1995. In general, the academy recommended: (1) limit risk to the average member of the critical group (starting point would be on the order of 2-20 mrem/year in terms of the dose/risk), (2) define reference biosphere and critical group characteristics by rule, (3) avoid quantitative subsystem requirements as in NRC's 10 CFR Part 60, (4) evaluate human intrusion separately (not possible to scientifically predict nature and timing of intrusion), and (5) conduct assessment for time of peak risk within the period of geologic stability, which for Yucca Mountain is on the order of 1-million years.

<u>2000 to 2009</u>

- EPA published site-specific standards for Yucca Mountain on June 13, 2001 (66 FR 32074). Final standards provided an individual dose limit of 15 mrem/year that is weighted by the probability and separate standards for the protection of ground water, both of which are specified for the initial 10,000 years. Evaluation of individual dose beyond 10,000 years is to be calculated and included in the Environmental Impact Statement; however, no regulatory standard applies. EPA also provided specific assumptions to be used in the performance assessments to be used for compliance with the standards (e.g., human intrusion scenario; reasonably, maximally exposed individual).

- NRC published site-specific regulations for Yucca Mountain on November 2, 2001 (66 FR 55732). NRC's regulations adopted the EPA standards but also included requirements for the performance assessment including the reference biosphere characteristics, and safety assessment for the preclosure or operational period. Additionally, NRC provided new requirements for multiple barriers that did not include quantitative subsystem requirements.

- NRC specified the value for probability of unlikely events in the regulations for Yucca Mountain (67 FR 62628, October 8, 2002), as directed by EPA in its final standards. After EPA's final standards were published, NRC elected to finalize its regulations and conduct separate rulemaking for specifying the probability for unlikely events (unlikely events are not considered in the assessments for groundwater protection and human intrusion). This rulemaking specified that unlikely features, events, and processes are those that are estimated to have less than one chance in 10 and at least one chance in 10,000 of occurring within 10,000 years of disposal.

- July 9, 2004 – the Court vacated EPA's standard for Yucca Mountain to the extent that it specified a 10,000-year compliance period (inconsistent with the NAS recommendations for a longer time such as 1-million years)

- June 8, 2008 - DOE submitted its license application to the NRC seeking authorization to construct a geologic repository at Yucca Mountain

- September 8, 2008 – NRC docketed the license application

- October 15, 2008 - EPA published revised standards for Yucca Mountain (73 FR 61256) that provided standards for the time period after 10,000 years to 1-million years. The revised standards specified a 100 mrem/year individual dose limit for individual protection and human intrusion and included limitations on the assessment for the period after 10,000 years (e.g., seismic analysis limited to drift damage, waste package failure, and water table rise; igneous analysis limited to waste package damage and release of radionuclides to the biosphere, atmosphere, or groundwater; general corrosion rate may use a constant representative corrosion rate; and climate change analysis is limited to the effects of increased water flow through the repository and that NRC shall specify in regulation the values to represent climate change such as temperature, precipitation, or infiltration rate of water).

- March 13, 2009 - NRC published revised regulations for Yucca Mountain (74 FR 10811) incorporating EPA's revised standards and specifying a deep percolation rate for the period after 10,000 years (i.e., constant-in-time deep percolation rate to be used to represent climate change varies between 10 and 100 mm/year [0.39 and 3.9 in./year]).

2010 to Present

- March 2010 - DOE filed a motion to withdraw its application before the Atomic Safety and Licensing Board. On September 30, 2010, DOE's Office of Civilian Radioactive Waste Management ceased operations and assigned its Yucca Mountain-related responsibilities to other offices within DOE.

- September 2011 - The Commission (with 4 members voting and 1 recusal) announced it was evenly divided on whether to overturn or uphold the Atomic Safety and Licensing Board denial of DOE's motion to withdraw the license application. The Commission directed the Atomic Safety and Licensing Board, in recognition of budgetary limitations, to complete all necessary and appropriate case management activities, and the Atomic Safety and Licensing Board suspended the proceeding on September 30, 2011.

- August 2013 - The U.S. Court of Appeals for the District of Columbia Circuit issued a decision granting a writ of mandamus and directed NRC to resume the licensing process for DOE's license application.

- November 2013 - The Commission directed the NRC staff to complete and issue the SER associated with the license application.

- January 2015 – The NRC staff completed all five volumes of the SER (Volume 1 entitled 'General Information' published August 2010, Volume 2 entitled 'Repository Safety before Permanent Closure' published January 2015, Volume 3 entitled 'Repository Safety after Permanent Closure' published October 2014, Volume 4 entitled 'Administrative and Programmatic Requirements, published December 2014, and Volume 5 entitled 'Proposed Conditions on the Construction Authorization and Probable Subjects of License Conditions' published January 2015).

- May 2016 – The NRC published its 'Supplement to DOE's Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada,' NUREG-2184. The document supplements environmental impact statements (EISs) the Department of Energy prepared on the proposed repository. DOE issued the final EIS in 2002, then supplemented it in June 2008 when it submitted a construction authorization application to the NRC. Under the Nuclear Waste Policy Act, the NRC is to adopt DOE's EIS to the extent practicable. The NRC staff recommended adoption of DOE's EISs in September 2008, but noted the need to supplement the study of groundwater effects in the Yucca Mountain aquifer beyond DOE's analyzed location at the site boundary. DOE ultimately deferred to the NRC to prepare the supplement.

All Nuclear Waste Fund activities at the NRC since 2011 have used funds remaining from previous years' appropriations.

2.1 References

Federal Register Notices

66 FR 32074 - 32135; June 13, 2001; 40 CFR Part 197 – Public Health and Environmental Radiation Standards for Yucca Mountain; NV; Final Rule; U.S. Environmental Protection Agency, Washington, D.C

66 FR 55732 - 55816; November 2, 2001; 10 CFR Part 63 – Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada; Final Rule; U.S. Nuclear Regulatory Commission; Washington, D.C.

67 FR 62629 - 62634; October 8, 2002; 10 CFR Part 63 – Specification of a Probability for Unlikely Features, Events and Processes; Final Rule; U.S. Nuclear Regulatory Commission; Washington, D.C.

73 FR 61256 61289; October 15, 2008; 40 CFR Part 197 – Public Health and Environmental Radiation Standards for Yucca Mountain, Nevada; Final Rule; U.S. Environmental Protection Agency, Washington, D.C

74 FR 10811 - 10830; March 13, 2009; 10 CFR Part 63 – Implementation of a Dose Standard After 10,000 Years; Final Rule; U.S. Nuclear Regulatory Commission; Washington, D.C.

Courts

D.C. Circuit of the United States Court of Appeals (No. 01-1258) (July 9, 2004) ML041980418

(Package: ML041980467)

Other References

NAS, 1995; Technical Basis for Yucca Mountain Standards; NAS Committee on Technical Bases for Yucca Mountain Standards; National Academy of Sciences (NAS); Washington, D.C.; August 1, 1995

3.0 Development of Regulations

NRC's generic regulations for high-level waste (HLW) disposal, which includes spent nuclear fuel were developed principally in the late 1970s and early 1980s. For example, the procedural requirements for licensing geologic HLW repositories under 10 CFR Part 60 were promulgated in 1981 (46 FR 13971; February 25, 1981), and technical criteria were promulgated in 1983 (48 FR 28194; June 21, 1983). In the early 1980s, the development of a performance assessment for evaluating geologic disposal was in its infancy and there were no quantitative evaluations available for understanding the performance of a potential repository site and design. Additionally, the U.S. Environmental Protection Agency (EPA) finalized its generic standards for HLW (40 CFR Part 191; 50 FR 38066; September 19, 1985).

NRC's generic regulations at 10 CFR Part 60 focused primarily on identifying a broad range of topics to address what might be relevant to a repository. The Commission also acknowledged at that time with respect to quantitative subsystem requirements (e.g., ground-water- travel time) that it was "appropriate to include reasonable generic requirements, that if satisfied, will ordinarily contribute to meeting the standards even though modifications may need to be made for some designs and locations" (48 FR 28196; June 21, 1983); and with respect to the level of detail for other prescriptive requirements of the design (e.g., shaft and borehole seals) that the emphasis should be placed on "the objectives that must be met and not become unduly concerned about the particular techniques that may be used in doing so" (48 FR 28198; June 21, 1983).

After the Yucca Mountain site was selected for further study as the HLW repository (Nuclear Waste Policy Amendments act of 1987), there were growing Congressional concerns that the U.S. Environmental Protection Agency (EPA) generic standards for HLW (40 CFR 191; 50 FR 38066; September 19, 1985) were not appropriately health based (see letter of J. Bennett Johnston to Robert W. Fri, May 20, 1993 provided in Appendix B "Congressional Mandate for This Report" of NAS Report). EPA's generic standards at 40 CFR 191 set an integrated release limit for specific radionuclides, such as carbon-14 (Table 1 in Appendix A in Final Standards [50 FR 38087; September 19, 1985) that raised questions about the relevance of this limit to health and safety. Thus, Congress took action to ensure the standards applicable to a repository at Yucca Mountain were appropriately health and safety based. In 1992, Congress directed EPA, at Section 801 of the Energy Policy Act of 1992, Public Law 102–486 (EnPA), to contract with the National Academy of Sciences (NAS) to advise EPA on the appropriate technical basis for public health and safety standards governing the Yucca Mountain repository. On August 1, 1995, the NAS Committee on Technical Bases for Yucca Mountain Standards issued its report, 'Technical Bases for Yucca Mountain Standards.' "In its report, NAS recommended an approach and content that is significantly different from that adopted by EPA for its disposal standards at 40 CFR 191 (no longer applicable to sites characterized under Section 113(a) of NWPA), as well as from that adopted by NRC for its existing generic regulations at Part 60." [64 FR 8641; February 22, 1999 – from NRC's proposed regulations for Yucca Mountain at 10 CFR Part 63]

The National Academy of Sciences published its recommendations for safety standards for Yucca Mountain that included both the safety limits but also the approach for estimating repository performance (NAS 1995). The recommendations included:

(a) <u>Postclosure safety limit</u>

Use a safety standard that sets a limit on the risk to individuals of adverse health effects from releases from the repository. A reasonable starting point for EPA's rulemaking would be 10^{-5} to 10^{-6} per year. (NAS 1995, pages 4-5)

(b) <u>Critical Group Approach</u>

Use a critical group approach to estimate doses or risks to avoid unreasonable assumptions regarding factors affecting dose and risk. The critical group has been defined by the International Commission on Radiological Protection (ICRP) as a relatively homogeneous group of people whose location and habits are such that they are representative of those individuals expected to receive the highest doses as a result of the discharges of radionuclides. (NAS 1995, pages 5-6)

(c) Exposure Scenarios

It is not possible to predict based on scientific analyses the societal factors required for an exposure scenario. Specifying exposure scenarios therefore requires a policy decision that is appropriately made in a rulemaking process conducted by EPA. (NAS 1995, pages 9-10)

(d) Compliance Period

Compliance assessment should be conducted for the time when the greatest risk occurs, within the limits imposed by long-term stability of the geologic environment, which is on the order of one million years. (NAS 1995, page 2 and pages 6-7)

(e) Human Intrusion

It is possible to carry out calculations of the consequences for particular types of intrusion events, for example drilling one or more boreholes into and through the repository. Calculations of this type might be informative in the sense that they can provide useful insight into the degree to which the ability of a repository to protect public health would be degraded by intrusion. EPA should specify in its standard a typical intrusion scenario to be analyzed for its consequences on the performance of the repository. The result of the analysis should not be integrated into an assessment of repository performance based on risk, but rather should be considered separately. The purpose of this consequence analysis is to evaluate the resilience of the repository to intrusion. (NAS 1995, pages 108 - 109)

EPA should require that the conditional risk resulting from the assumed intrusion scenario be no greater that the risk limits adopted for the undisturbed repository case. (NAS 1995, page 121)

At this time, NRC re-examined its generic regulations at 10 CFR Part 60 based both on EnPA and the NAS report on the Technical Basis for Yucca Mountain Standards. As discussed in the proposed Part 63 (64 FR 8640; February 22, 1999):

> "The Commission considered the most direct and time-efficient approach to the specification of concise, site specific criteria for Yucca Mountain that are consistent with current assumptions, with site-specific information and performance assessment experience, and with forthcoming EPA standards would be to develop site-specific regulations that apply solely to Yucca Mountain. In establishing these criteria, the Commission sought to establish a coherent body of risk-informed, performance-based criteria for Yucca Mountain compatible with the Commission's overall philosophy of risk-informed, performance-based regulation. "Stated succinctly, risk-informed, performance-based regulation is an approach in which risk insights, engineering analysis and judgment (e.g., defense in depth), and performance history are used to (1) focus attention on the most important activities, (2) establish objective criteria for evaluating performance, (3) develop measurable or calculable parameters for monitoring system and licensee performance, (4) provide flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes, and (5) focus on the results as the primary basis for regulatory decision-making." (64 FR 8643; February 22, 1999)

NRC's site-specific regulations at 10 CFR Part 63 are generally the same as the generic regulations at 10 CFR Part 60 with respect to the procedural aspects of the regulation; however, the technical criteria were revised to take full advantage of a risk-informed, performance-based approach (SECY 98-300; NRC 1997). The technical criteria in Part 63 included requirements for the postclosure and preclosure safety assessments and the safety limits (i.e., dose limits) and did not include other criteria in the generic regulations (e.g., quantitative subsystem criteria for the postclosure barrier performance, design criteria) that were considered unnecessary in light of the information that would be evaluated in the overall safety analyses. Additionally, certain requirements in Part 63 reflect characteristics of the Yucca Mountain site consistent with the NAS recommendations and EPA's site-specific standards at 40 CFR Part 197, including the specification of a compliance period of 1-million years.[1]

Appendix A "Development of Regulations for Geologic Disposal of High Level Waste and Spent Fuel" provides further discussion on the development and approach for the rulemaking primarily for key aspects of the postclosure performance assessment.

3.1 Site Specific Regulations for Yucca Mountain (10 CFR Part 63)

NRC's regulations at 10 CFR Part 63 provide site-specific criteria for geologic disposal at Yucca Mountain. Pursuant to 10 CFR Part 63, there are several stages in the licensing process: the site characterization stage, the construction stage, a period of operations, and termination of the license. The multi-staged licensing process affords the Commission the flexibility to make decisions in a logical time sequence that accounts for DOE collecting and analyzing additional information over the construction and operational phases of the repository. The period of operations includes (i) the time during which emplacement would occur, (ii) any subsequent period before permanent closure during which the emplaced wastes are retrievable, and (iii) permanent closure. Permanent closure represents the end of the performance confirmation program; final backfilling of the underground facility, if appropriate; and the sealing of shafts, ramps, and boreholes. The postclosure period, as the name implies, follows permanent closure and is the time period when the repository barriers are expected to provide passive safety functions (i.e., maintenance and human intervention is not required) are relied on due to the long postclosure period (i.e., up to 1-million years). NRC's regulations provide requirements for postclosure monitoring and institutional controls, however, they are not relied on in the postclosure safety assessment for demonstration of compliance with the dose limits.

A brief description of key aspects of the technical criteria in Part 63 are described below for the (1) preclosure safety assessment and limits, (2) performance confirmation program and retrievability, (3) postclosure safety assessment and limits for the initial 10,000 years, (4) post-closure safety assessment and limits for the period after 10,000 years, (5) multiple barrier requirement.

3.1.1 Preclosure Safety Assessment and Limits

The period of operations includes (i) the time during which emplacement would occur, (ii) any subsequent period before permanent closure during which the emplaced wastes are retrievable, and (iii) permanent closure. The regulations in 10 CFR Part 63 (Subparts E and K) provide performance objectives and requirements that the site and design must comply with for the operational period.

> "In accordance with 10 CFR 63.21, the applicant must include in its SAR a preclosure safety analysis (PCSA). As described in 10 CFR 63.102(f), the PCSA identifies and categorizes event sequences and identifies structures, systems, and components (SSCs) important to safety (ITS) and associated design

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[1] After publication of final EPA standards (66 FR 32074; June 13, 2001) and NRC regulations (66 FR 55732; November 2, 2001) lawsuits were filed on a variety of aspects. The U.S. Court of Appeals for the District of Columbia Circuit upheld both EPA's standards and NRC's regulations on all but one of the issues raised by the petitioners – the court disagreed with EPA's decision to adopt a 10,000-year period for compliance in contradiction to the NAS recommendations specifying a compliance period on the order of 1-million years. Subsequently EPA specified additional standards (73 FR 61256; October 15, 2008) and NRC specified requirements (74 FR 10811; March 13, 2009) that included both safety limits and specific requirements for the safety assessment for the period after 10,000 years - there is a pending legal challenge to the final standards and regulations for the period after 10,000 years; this case is, however, being held in abeyance, subject to periodic status reports. The case may resume should Yucca Mountain program activities resume.

bases and criteria. The PCSA is part of the risk-informed and performance-based review, which is described further in the following section. An event sequence, as defined in 10 CFR 63.2, is a series of actions and/or occurrences within the natural and engineered components of the facility that could potentially expose individuals to radiation. The applicant's PCSA must demonstrate that the repository, as proposed to be designed, constructed, and operated, will meet the specified radiological dose limits throughout the preclosure period. The applicant must also demonstrate that the GROA design will not preclude retrievability of the wastes, in whole or in part, from the underground facility where these wastes will be emplaced for permanent disposal (10 CFR 63.111)." (SER Vol. 2 page xxi)

As explained at §63.102(f) (Preclosure Safety Analysis):

> "The preclosure safety analysis is a systematic examination of the site; the design; and the potential hazards, initiating events and their resulting event sequences and potential radiological exposures to workers and the public. Initiating events are to be considered for inclusion in the preclosure safety analysis for determining event sequences only if they are reasonable (i.e., based on the characteristics of the geologic setting and the human environment, and consistent with precedents adopted for nuclear facilities with comparable or higher risks to workers and the public). The analysis identifies structures, systems, and components important to safety."

Part 63 implemented the risk-informed, performance-based approach that provided for dose limits according to the likelihood of the exposure and specific requirements for the applicant's preclosure safety analysis that is used to demonstrate compliance with the dose limits. Some key aspects of 10 CFR Part 63 are:

(1) Dose limit of 0.15 mSv (15 mrem) per year of that applies during normal operations and category 1 event sequences (i.e., those events that are expected to occur one or more times before permanent closure) for any real member of the public located beyond the boundary of the site under 'Protection against radiation exposures and releases of radioactive material' [§ 63.111(a)]

(2) Dose limit of 0.05 Sv (5 rem) from a single category 2 event sequence (i.e., a sequence that has at least one chance in 10,000 of occurring before permanent closure) for an individual located on, or beyond, any point on the boundary of the site under the 'Numerical Guides for design objectives' [§ 63.111(b)]

(3) Requirements for the preclosure safety analysis of the geologic repository operations area [GROA] (i.e., the high-level radioactive waste facility that is part of a geologic repository, including both surface and subsurface areas, where waste handling activities are conducted) that is used to demonstrate the dose limits will be met [§§ 63.111(c) and 63.112]

3.1.2 Performance Confirmation and Retrievability

NRC's regulations for disposal of HLW (both the generic regulations at 10 CFR Part 60 and the site-specific regulations for Yucca Mountain at 10 CFR Part 63) were developed with a recognition that the period of development and emplacement of waste would be lengthy (i.e., decades) and this time period could appropriately be used to conduct further studies and analyses to provide additional confirmation that the repository postclosure barriers would perform consistent with the models and tests supporting the earlier safety decisions.

NRC's site-specific regulations at 10 CFR Part 63 for Yucca Mountain describes the relationship between the performance confirmation program and the performance objectives for the geologic repository after permanent closure (§ 63.113) at § 63.102(m) [Performance Confirmation]:

> A performance confirmation program will be conducted to evaluate the adequacy of assumptions, data, and analyses that led to the findings that permitted construction of the repository and subsequent emplacement of the wastes. Key geotechnical and design parameters, including any interactions between natural and engineered systems and components, will be monitored throughout site characterization, construction, emplacement, and operation to identify any significant changes in the conditions assumed in the license application that may affect compliance with the performance objectives specified at § 63.113(b) and (c).

As described at § 63.102(c), the performance confirmation program is conducted throughout the stages in the licensing process for a geologic repository:

> There are several stages in the licensing process. The site characterization stage when the performance confirmation program is started, begins before submission of a license application, and may result in consequences requiring evaluation in the license review. The construction stage would follow the issuance of a construction authorization. A period of operations follows the Commission's issuance of a license. The period of operations includes the time during which emplacement of wastes occurs; any subsequent period before permanent closure during which the emplaced wastes are retrievable; and permanent closure, which includes sealing openings to the repository. Permanent closure represents the end of the performance confirmation program; final backfilling of the underground facility, if appropriate; and the sealing of shafts, ramps, and boreholes.

It is anticipated that issues and concerns would be discussed throughout conduct of the performance confirmation program consistent with and appropriate to the significance of the issue/concern and the specific stage of the licensing process at time the issue/concern is raised (e.g., § 63.73 "Reports of deficiencies" requires DOE to promptly notify the Commission of significant deviations from design criteria and conditions in the construction authorization or license).

The multi-staged licensing process affords the Commission the flexibility to make decisions in a logical time sequence that accounts for DOE collecting and analyzing additional information over the construction and operational phases of the repository. A final review of the information from the performance confirmation program would be made at time to amend the License for permanent closure, as described at § 63.51:

> An update of the assessment of the performance of the geologic repository for the period after permanent closure. The updated assessment must include any performance confirmation data collected under the program required by subpart F, and pertinent to compliance with § 63.113.

The Commission is afforded the flexibility in making its safety decision throughout the long operational period, while the performance confirmation program continues to collect information, by requiring DOE to maintain the ability to retrieve waste throughout this time period as stated at § 63.111 Performance objectives for the geologic repository operations area through permanent closure:

> The geologic repository operations area must be designed to preserve the option of waste retrieval throughout the period during which wastes are being emplaced and thereafter, until the completion of a performance confirmation program and Commission review of the information obtained from such a program. [§ 63.111(e)(1)]

Following permanent closure and the decontamination or decontamination and dismantlement of surface facilities at the Yucca Mountain site, DOE may apply for an amendment to terminate the license (§ 63.52 – Termination of License). Termination results in a single government agency (i.e. DOE) responsible for oversight of the repository over the postclosure period when the repository barriers are intended to provide passive safety (i.e., human intervention and maintenance is not anticipated nor required). DOE would be responsible for monitoring and oversight as required at § 63.51 (license amendment for permanent closure) that includes requirements for programs for post-permanent closure monitoring of the geologic repository [§ 63.51(a)(2) and measures to prevent activities that could impair the long-term isolation of emplaced waste within the geologic repository [§ 63.51(a)(3)].

3.1.3 Postclosure Safety Assessment for the Initial 10,000 Years

In finalizing the risk-informed, performance-based regulations at 10 CFR Part 63; the Commission stated:

> "…a performance assessment, developed with sufficient credibility, is the best means to provide useful information to the Commission for making an informed, reasonable licensing decision. The Commission recognizes, however, the uncertainties inherent in evaluating a first-of-a-kind facility like the repository and in estimating system performance over very long time periods (i.e., 10,000 years). Thus, proposed part 63 contained requirements to ensure that: (1) uncertainties inherent in any performance assessment are thoroughly articulated and analyzed or addressed; (2) DOE's performance assessment is tested (corroborated) to the extent practicable; and (3) there are additional bases, beyond the performance assessment, that provide confidence that the postclosure performance objectives will be met." (66 FR 55747; November 2, 2001)

The postclosure safety assessment (referred to as a performance assessment), as described in Part 63, is a systematic analysis that identifies the features, events, and processes (i.e., specific conditions or attributes of the geologic setting, degradation, deterioration, or alteration processes of engineered barriers, and interactions between the natural and engineered barriers) that might affect performance of the geologic repository; examines their effects on performance; and estimates the radiological exposures to the reasonably maximally exposed individual [§ 63.101(j)].

The NAS, as described above, made recommendations for the postclosure safety limit and the approach or performance assessment used to estimate the radiological exposures. Thus, EPA's standards (which are to be incorporated into NRC's regulations) and NRC's regulations reflect, as relevant and appropriate, the NAS recommendations. Key aspects of NRC's requirements and the incorporated EPA standards for the postclosure safety assessment and dose limits for the initial 10,000 years are:

(1) §§ 63.311(a)(1) and 63.321(b)(1) specify a dose limit of 0.15 mSv (15 mrem) per year of that applies to the RMEI for the undisturbed repository (i.e., no human intrusion occurs) and also for the separate estimate human intrusion scenario (human intrusion scenario is defined at § 63.322).

(2) EPA's separate standards for protection of groundwater that provide separate limits for individual radionuclides were incorporated into the regulations at § 63.331.

(3) Requirements for the performance assessment (PA) were specified at § 63.114 that provides requirements for data and support for the parameters and models in the PA including accounting for uncertainty.

(4) Probability limits for the features, events, and processes that need to be considered for inclusion in the PA are specified at § 63.342(a) and state that very unlikely FEPs are not required to be considered in the PA (i.e., those that are estimated to have less than one chance in 100,000,000 per year of occurring). Additionally, the PA used to demonstrate compliance with the human intrusion dose limit and the separate standards for groundwater protection are not required to include unlikely FEPs (i.e., those that are estimated to have less than one chance in 100,000 per year of occurring and at least one chance in 100,000,000 per year of occurring) as specified at § 63.342(b).

(5) The required characteristics of the reference biosphere to be used in the PA are specified at § 63.305.

(6) The required characteristics of the RMEI to be used in the PA for estimating radiological exposures are specified at § 63.312.

3.1.4 Postclosure Safety Assessment for the Period after 10,000 Years

After publication of final EPA standards (66 FR 32074; June 13, 2001) and NRC regulations (66 FR 55732; November 2, 2001) lawsuits were filed on a variety of aspects, including NRC's removal of quantitative sub-system requirements. The U.S. Court of Appeals for the District of Columbia Circuit upheld both EPA's standards and NRC's regulations on all but one of the issues raised by the petitioners – the court disagreed with EPA's decision to adopt a 10,000-year period for compliance with the standards and NRC's adoption of that 10,000-year compliance period in NRC's implementing regulations (No. 01-1258) [July 9, 2004]. The court concluded, in part:

> V. Conclusion
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> "In sum, we vacate 40 C.F.R. part 197 to the extent that it incorporates a 10,000-year compliance period because, contrary to EnPA section 801(a), that compliance period is not ''based upon and consistent with'' the recommendations of the National Academy of Sciences. The remaining challenges to the EPA rule are without merit. We vacate the NRC rule insofar as it incorporates EPA's 10,000-year compliance period. In all other respects, we deny Nevada's petition for review challenging the NRC rule."

Following this decision, EPA and NRC evaluated revisions to the standards and regulations to accommodate implementation of an unprecedented time period of 1-million years for evaluating the postclosure safety of Yucca Mountain as recommended by the NAS. In short, the NAS in its report on the Technical Basis for Yucca Mountain Standards recommended that "compliance assessment be conducted for the time when the greatest risk occurs, within the limits imposed by long-term stability of the geologic environment" (page 7), and that the time scale of the long-term stability of the fundamental geologic regime at Yucca Mountain[2] is a time scale that is on the order of 1-million years (page 6).

The EPA and NRC recognized that, with such an unprecedented time frame as 1-million years, they needed to carefully consider how to reasonably implement such a long compliance period. EPA in its final standards for the period after 10,000 years (73 FR 61256; October 15, 2008) choose to set a separate dose for the period after 10,000 years up to 1-million years as well as provide additional limits for the PA for this time period due to consideration that the level of uncertainty increases as the time period covered by DOE's performance assessment increases (73 FR 61260 and 61261; October 15, 2008). Key aspects of NRC's requirements for the period after 10,000 years are:

(1) §§ 63.311(a)(2) and 63.321(b)(2) specify a dose limit for the period after 10,000 years of 1.0 mSv (100 mrem) per year of that applies to the RMEI for the undisturbed repository (i.e., no human intrusion occurs) and also for the separate estimate human intrusion scenario (i.e., human intrusion scenario is defined at § 63.322).

(2) Requirements for the performance assessment (PA) for the period after 10,000 years were specified at § 63.114(b) that states performance assessment methods used to satisfy the requirements for the PA for the initial 10,000 years are considered sufficient for the PA for the period after 10,000 years and through the period of geologic stability (defined to end at 1-million years after disposal at § 63.302).

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[2] It should be noted that the NAS specifically identified the period of geological stability was the driver for a 1-million year compliance period. If similar logic were to be applied at a different site where the period of geological stability was determined to be significantly longer (e.g., 10-million years) it could imply the need for a compliance period of 10 million years at such a site if peak dose is estimated to occur around the 10 million year period.

(4) Probability limits for the features, events, and processes that need to be considered for inclusion in the PA specified at § 63.342 for the initial 10,000 years are also applicable to the period after 10,000 years.

(5) The PA for the period after 10,000 years is required to project the continued effects of the FEPs as included in the PA for the initial 10,000 years, however, specific limits for representing the seismic analysis, igneous analysis, effects of climate change (i.e., specification of a deep percolation rate), and the effects of general corrosion on the engineered barriers to be used during the period after 10,000 years [§ 63.342(c)].

(6) The required characteristics of the reference biosphere to be used in the PA are specified at § 63.305.

(7) The required characteristics of the RMEI to be used in the PA for estimating radiological exposures are specified at § 63.312.

Further discussion of the requirements in Part 63 for the period after 10,000 years, including the technical basis and support for the specification of a deep percolation rate for representing the effects of climate change, is provided in Appendix A.

3.1.5 Multiple Barrier Requirement

Development of safety regulations for HLW geologic disposal recognized from the beginning the unique challenges of providing safety over times of thousands of years after a repository is permanently closed (i.e., the postclosure period). Discussion of the role and need of both passive (i.e., not requiring maintenance or human actions) and multiple (i.e., engineered and geologic) barriers for ensuring postclosure safety has been discussed throughout the past five decades (e.g., NRC 1983, ICRP 1998, NEA 1999, POSIVA 2007, and IAEA 2011). Although multiple barriers are a part of every HLW geologic repository program internationally and nationally, the NRC also imposed specific quantitative limits for three specific subsystems associated with a barrier's performance (i.e., longevity of the waste package, release rate from the waste form, and travel time of radionuclides in groundwater) when it developed its generic regulations at 10 CFR Part 60 (48 FR 28224; June 21, 1983). NRC described its growing concerns with the quantitative subsystem requirements in Part 60 when it proposed site-specific regulations for Yucca Mountain at Part 63 and acknowledged support for risk-informed, performance-based regulations without the imposition of quantitative subsystem requirements. Based, in part, on the considerable evolution in the technical methods for assessing the performance of a geologic repository, the Commission proposed a different approach for the multiple barrier requirements in 10 CFR Part 63. NRC finalized its approach for multiple barriers in 10 CFR Part 63 on November 2, 2001 (66 FR 55732) requiring that (1) the geologic repository must include multiple barriers, consisting of both natural barriers and an engineered barrier system (§ 63.113(a)) and the Department of Energy (DOE) must (a) Identify the design features of the engineered barrier system, and the natural features of the geologic setting, that are considered barriers important to waste isolation, (b) Describe the capability of barriers, identified as important to waste isolation, taking into account uncertainties in characterizing and modeling the behavior of the barriers, and (c) Provide the technical basis for the description of the capability of barriers, identified as important to waste isolation (the technical basis for each barrier's capability shall be based on and consistent with the technical basis for the performance assessment) (§ 63.115).

Further discussion of NRC's technical and basis for support for the multiple barrier requirements in Part 63 is provided in Appendix A.

3.2 Summary

The regulatory section provides a high-level view of the overall approach for regulations for a geologic repository at Yucca Mountain. Appendix A provides further details regarding the development and support for aspects that are most unique to a deep geologic repository (e.g., postclosure safety and multiple barrier approach for the postclosure period). Other aspects of the regulations such as operational safety for surface facilities that receive and handle waste packages are similar to handling of canisters of spent fuel at other NRC regulated facilities (i.e., operating reactors and storage facilities) and further discussion is not provided in Appendix A. Further discussion for the site specific regulations at 10 CFR Part 63 is provided in the Statements of Consideration for the Part 63 for the initial 10,000 years (66 FR 55732; November 2, 2001), specification of a probability for unlikely features, events and processes (67 FR 62628; October 8, 2002) and implementation of dose standard after 10,000 years (74 FR 10811; March 13, 2009). Additionally, the Yucca Mountain Review Plan (YMRP) [NRC 2003] provides additional insights regarding the regulatory requirements. [need to identify ISGs]

3.3 References

Federal Register Notices

46 FR 13971 - 13987; February 25, 1981; 10 CFR Part 60 - Disposal of High-Level Radioactive Wastes in Geologic Repositories: Licensing Procedures – Final Rule; U.S. Nuclear Regulatory Commission; Washington, D.C.

48 FR 28194 - 28229; June 21, 1983; 10 CFR Part 60 - Disposal of High-Level Radioactive Wastes in Geologic Repositories Technical Criteria – Final Rule; U.S. Nuclear Regulatory Commission; Washington, D.C.

50 FR 38066 - 38089; September 19, 1985; 40 CFR Part 191 – Environmental Standards for the Management and Disposal of Spent Nuclear Fuel; High-Level and Transuranic Radioactive Wastes – Final Rule; U.S. Environmental Protection Agency, Washington, D.C.

64 FR 8640 - 8679; February 22, 1999; 10 CFR Part 63 – Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada; Proposed Rule; U.S. Nuclear Regulatory Commission; Washington, D.C.

66 FR 32074 - 32135; June 13, 2001; 40 CFR Part 197 – Public Health and Environmental Radiation Standards for Yucca Mountain; NV; Final Rule; U.S. Environmental Protection Agency, Washington, D.C

66 FR 55732 - 55816; November 2, 2001; 10 CFR Part 63 – Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada; Final Rule; U.S. Nuclear Regulatory Commission; Washington, D.C.

67 FR 62629 - 62634; October 8, 2002; 10 CFR Part 63 – Specification of a Probability for Unlikely Features, Events and Processes; Final Rule; U.S. Nuclear Regulatory Commission; Washington, D.C.

73 FR 61256 61289; October 15, 2008; 40 CFR Part 197 – Public Health and Environmental Radiation Standards for Yucca Mountain, Nevada; Final Rule; U.S. Environmental Protection Agency, Washington, D.C

74 FR 10811 - 10830; March 13, 2009; 10 CFR Part 63 – Implementation of a Dose Standard After 10,000 Years; Final Rule; U.S. Nuclear Regulatory Commission; Washington, D.C.

Courts

D.C. Circuit of the United States Court of Appeals (No. 01-1258) (July 9, 2004) ML041980418

(Package: ML041980467)

Other References

IAEA (2011), Disposal of Radioactive Waste, Specific Safety Requirements No. SSR-5. International Atomic Energy Agency, Vienna.

ICRP (1998), Radiation Protection Recommendations as Applied to the Disposal of Long-lived Solid Radioactive Waste, ICRP Publication 81. Ann. ICRP 28(4), Pergamon Press, Oxford.

NEA (1999), “Confidence in the Long-term Safety of Deep Geological Repositories, Its Development and Communication”, OECD Publishing, Paris.

POSIVA (2007), “Safety Assessment for a KBS-3H Spent Nuclear Fuel Repository at Olkiluoto”, POSIVA 2007-09, Posiva Oy, Eurajoki, Finland.

NAS, 1995; Technical Basis for Yucca Mountain Standards; NAS Committee on Technical Bases for Yucca Mountain Standards; National Academy of Sciences (NAS); Washington, D.C.; August 1, 1995

NRC, 2003. NUREG–1804, “Yucca Mountain Review Plan—Final Report,” Rev. 2, ML032030389; U.S. Nuclear Regulatory Commission; Washington, D.C.; Washington, DC; July, 2003.

NRC, 1997, “Policy Issue (Notation Vote); Subject: Proposed Strategy for Development of Regulations Governing Disposal of High-Level Radioactive Wastes in a Proposed Repository at Yucca Mountain, Nevada,” from: L. Joseph Callan (EDO) for the Commissioners, December 24, 1997, SECY-97-300; ADAMS# ML032830444.

4.0 NRC's Review Preparations

The NRC conducted a comprehensive range of activities in preparation to perform a technical review of the DOE license application for geologic disposal at a potential repository at Yucca Mountain. These activities included independent technical evaluations at NRC and the Center for Nuclear Waste Regulatory Analyses (CNWRA) that included both laboratory and field investigations, development of a preclosure safety analysis capability, development of a performance assessment capability to independently evaluate repository performance following permanent closure, development of a Yucca Mountain Review plan including Interim Staff Guidance documents (ISGs); participation in numerous public technical exchanges between NRC and DOE that began in the 1980s; and public interactions with stakeholders.

In addition to the public interactions held between DOE and NRC, the NRC conducted a variety of stakeholder engagement activities during pre-licensing and during the license application review. One primary purpose of the NRC's outreach was to explain, in an open and transparent way, (i) the NRC's role in the licensing process, as well as context about the roles of other agencies and parties; (ii) the license application review and licensing processes; and (iii) the confirmatory studies and review activities NRC was undertaking to ensure an independent, robust evaluation. NRC also sought to build relationships with stakeholders and gather information from them to enhance the staff's understanding of participants' concerns. The public outreach team was drawn from NRC and CNWRA staff members engaged in other Yucca Mountain-related technical projects who could contribute to public outreach. The NRC held its first Yucca Mountain-related public meetings in 1999 in Nevada on the topic of the proposed 10 CFR Part 63 rule. Public meetings continued over the duration of the program and ranged from small group discussions with specific stakeholders (such as Tribal groups), to open houses in rural community centers, to large public meetings in Las Vegas. NRC staff held special workshops with parties that had received official designation of Affected Units of Local Government (AULGs). The workshops were designed to provide targeted information about the project and licensing process so that the parties could meaningfully participate in the licensing proceedings.

To facilitate information sharing, the staff developed materials using a broad range of media. Outreach materials included presentations, posters, flyers, fact sheets, contact cards, website materials, and videos. Staff hosted a booth, gave presentations, and developed posters for professional conferences. A tabletop three-dimensional (3D) model of a cross section of Yucca Mountain – including light-up sections with recorded voice descriptions – was created as a visualization tool. Each of these products, together with the staff-stakeholder interactions afforded by public meetings and workshops, contributed to NRC's goal to conduct its licensing review in a transparent, understandable way. These and other public outreach media and activities were recorded in a series of reports (Juckett, 2010, 2011a; 2011b). Public outreach activities ceased upon suspension of the license review, but the NRC and CNWRA staff again engaged stakeholders during development of the Supplement to the DOE's EIS, as discussed in Section 7.

An archive of meetings that includes these public outreach activities and NRC/DOE interactions may be found at https://www.nrc.gov/waste/hlw-disposal/historical-information/public-involvement/mtg-archive.html.

4.1 Independent Technical Capabilities

A key aspect of evaluating compliance with NRC's regulations and review of geologic disposal is the development of an independent performance assessment capability. NRC demonstrated its initial capability to conduct a performance assessment in NUREG-1327 "Initial Demonstration of the NRC's Capability to Conduct a Performance Assessment for a High-Level Waste Repository" (NRC 1992). Early work on this independent capability also helped guide development of the regulations and its technical basis. The NRC continued to develop its performance assessment capability with the assistance of its contractors at the Center for Nuclear Waste Regulatory Analyses (CNWRA). The NRC used the term 'iterative performance assessment" to characterize its program that continued to conduct assessments of the Yucca Mountain repository making use of available design and site information to perform sensitivity and uncertainty analyses to identify areas for additional model development and data collection (e.g., NRC 1995, NRC 2001, CNWRA 2004).

It is important to recognize that any performance assessment is based on the scientific information and data that supports its models and parameters. Thus, the development of scientific information and the uncertainty and sensitivity analyses supporting the performance assessment were conducted collaboratively to assist the staff's understanding of risk-significant issues associated with the potential repository at Yucca Mountain. Two significant examples of the staff using its technical capabilities in data and uncertainty analyses, process model development, and performance assessment to guide its preparations to review the license application are:

1) From FINAL STAFF RESPONSE TO MARCH 19, 2003, STAFF REQUIREMENTS MEMORANDUM ON THE WASTE ARENA BRIEFING (NRC 2003):

"Following this briefing, the Commission directed the staff, in the subject Staff Requirements Memorandum (SRM), to provide a report documenting the status of Key Technical Issue (KTI) agreements and report on the risk significance ranking of the 293 KTI agreements.
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Based on its understanding of current performance assessments, the staff rated the 293 KTI agreements according to their risk significance. The staff judged risk significance by evaluating the impact the requested information could have on current risk estimates and uncertainties in the risk estimates, taking into account the performance of multiple barriers (i.e., defense-in-depth). In a second step, the staff evaluated the technical difficulty of each agreement, and assessed the staff resources that would be required to evaluate the associated DOE responses.
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Generally, high-risk significance during the post-closure period is associated with features, events, and processes that could affect a large number of waste packages or significantly affect the releases from the waste package, or significantly affect the transport of radionuclides through the geosphere. Using this criterion, the following six areas have the highest significance for estimating performance: (1) corrosion of the drip shield and waste package, including the chemistry of water contacting the drip shield and waste package; (2) mechanical degradation of the drip shield and waste package caused by the long-term degradation of repository drifts; (3) effects of in-package chemistry on the dissolution of the waste form; (4) radionuclide transport in the saturated zone; (5) probability of volcanic disruption of the repository; and (6) entrainment and transport of radionuclides in volcanic ash. Thus, agreements that provide the technical basis supporting DOE's understanding and representation of the proposed repository in these six areas are ranked as high-risk significant. For example, results from testing of the waste package materials under representative repository conditions and evaluation of aeromagnetic data to determine the probability of volcanic activity would be ranked as high-risk-significant agreements.

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In addition to the six areas of high-risk significance for the post-closure performance, two other areas were identified as high-risk significant. First, development of confidence in the model abstractions used in the performance assessment was ranked as high-risk significance. Agreements related to DOE's evaluation of the degree of realism and conservatism in the models, and the representation of uncertainty in the models were ranked as high-risk significance. Second, the consideration of accidental aircraft crashes during the operational or pre-closure phase of the repository was ranked as high-risk significant. Based on this understanding of risk significance, the agreements were categorized as 41 of high-risk significance, 92 of medium-risk significance, and 160 of low-risk significance (see Attachment 1 for details on the status and risk ranking of the agreements).

The risk insights provided in this memorandum are part of a larger effort referred to as the High-Level Waste Risk Insights Initiative. As part of this initiative, staff has developed an integrated synopsis of its current understanding of key issues in repository performance. This risk baseline information is provided in Attachment 2. The baseline will be updated as appropriate to address changes in DOE's proposed repository design and modeling approach. We plan to brief the Advisory Committee on Nuclear Waste during its public meeting in June 2003 and address any recommendations as we complete the initiative report by October 2003. The risk baseline will also be updated prior to receipt of the license application." (NRC, 2003)

2) From NUREG-1762 Integrated Issue Resolution Status Report (NRC 2005):

"This report provides an overview of available information and status (as of March 2004, with exceptions as noted) of the Key Technical Issue agreements reached between DOE and NRC. The report also documents the risk insights (Appendix D) and information considered by the NRC staff in formulating their views, including the results of in-depth reviews of available DOE and contractor documents; the independent confirmatory work of NRC and its contractor, the Center for Nuclear Waste Regulatory Analyses; published literature; and other publicly available information." (page iii)

"Starting in August 2000, the DOE and NRC staffs conducted technical exchanges with the specific objective of prelicensing issue resolution of what were identified as the key technical issues. The technical exchanges were held as open public meetings. Available information was evaluated for its sufficiency for inclusion in any license application. Where such information was determined to be insufficient, NRC reached agreements with DOE to provide further information or analyses. These agreements specify the additional information DOE will collect, a schedule for obtaining such information, and a mechanism for providing the information to the NRC staff. The key technical issues are defined as resolved at the staff level when the NRC staff considers the information gathered by DOE sufficient for the staff to conduct a detailed technical review after submittal of a potential license application. Resolution, however, does not imply any conclusions regarding the end result of such a review, and any issue can be reopened if new information becomes available." Page 8-1

"As part of an ongoing effort to increase the use of risk information in its regulatory activities, the U.S. Nuclear Regulatory Commission (NRC) High-Level Waste Program is enhancing documentation of risk information and synthesizing the information to better support a risk-informed regulatory program. This-effort is referred to as the Risk Insights Initiative. This report documents the results of the Risk Insights Initiative and provides the results in the form of the Risk Insights Baseline Report. The Risk Insights Baseline Report serves as a common reference for staff to use in risk-informing the NRC High-Level Waste Program, as it continues through prelicensing regulatory activities and prepares to review a license application that may be submitted by the U.S. Department of Energy (DOE) for a potential high-level waste repository at Yucca Mountain, Nevada.

The risk insights presented in this report address the staff current understanding of the repository system following cessation of repository operations and permanent closure of the repository through the 10,000-year compliance period (i.e., the postclosure period). The risk insights are drawn from the staff experience gained through the development and exercise of the Total-System Performance Assessment (TPA) computer code, technical analyses conducted by the staff to support prelicensing interactions with DOE, and analyses conducted by DOE and others. If DOE submits a license application for a potential repository at Yucca Mountain, staff will review the information provided by DOE and make its determinations based on information available at that time." (Appendix D page D-ix)

To help prepare for preclosure safety reviews, CNWRA developed an independent PCSA tool (Maxwell et al., 2005). The tool was essential to providing the NRC and CNWRA staffs experience and a better understanding of how DOE would demonstrate preclosure safety in its license application. It was designed to support the review methodology laid out in the YMRP that involves evaluation of: (i) site description; (ii) description and design of structures, systems, and components, equipment, and operational process activities; (iii) identification of hazards and initiating events; (iv) identification of event sequences; and (v) consequence analysis. Key outputs are a consequence analysis (i.e., radiological doses to workers and the public), safety assessment (i.e., integration of results for comparison with dose compliance objectives), and risk assessment. The PCSA tool was used for confirmatory analyses, aided evaluations of the safety significance of structures, systems, and components, and provided risk insights. In addition, the PCSA Tool was intended to support the staff in inspection activities and assessments of regulatory safety during operations.

There were also hundreds of NRC and Center for Nuclear Waste Regulatory Analyses (CNWRA) reports over the two decades prior to DOE submitting its license application in 2008 that documented both technical analyses and the results of independent laboratory and field investigations. Appendix B (Postclosure Safety Review for a Potential Repository at Yucca Mountain) provides key aspects of NRC's review and identifies a number of the NRC and CNWRA reports that assisted NRC's review. A depository of knowledge management reports that document NRC and CNWRA prelicensing independent technical activities may be found at https://www.nrc.gov/waste/hlw-disposal/historical-information#km.

4.2 Yucca Mountain Review Plan (YMRP)

"The Yucca Mountain Review Plan is guidance to the U.S. Nuclear Regulatory Commission staff for review of any license application from the U.S. Department of Energy for a geologic repository for disposal of high-level radioactive waste at Yucca Mountain, Nevada. The U.S. Nuclear Regulatory Commission has directed the staff to carry out risk-informed, performance-based regulatory programs. 10 CFR Part 63 is risk-informed and performance-based, because risk of health effects to the reasonably maximally exposed individual is the basis for its performance objectives. 10 CFR Part 63 also requires protection of ground water by limiting the radioactivity in a representative volume of ground water and an assessment of repository performance under conditions of human intrusion. The U.S. Nuclear Regulatory Commission will base its licensing decision on whether the U.S. Department of Energy has demonstrated compliance with the performance objectives. Therefore, the Yucca Mountain Review Plan is risk-informed and performance-based. The principal purpose of the Yucca Mountain Review Plan is to ensure the quality and uniformity of U.S. Nuclear Regulatory Commission staff licensing reviews. Yucca Mountain Review Plan sections present the areas of review, review methods, acceptance criteria, evaluation findings, and references the staff will use for its review. There are sections for reviews of general information, repository safety before permanent closure, repository safety after permanent closure, the research and development program to resolve safety questions, the performance confirmation program, and administrative and programmatic requirements." (NRC 2003, page xv)

"10 CFR Part 63 requires the U.S. Department of Energy to conduct a performance assessment to demonstrate compliance with postclosure performance objectives. A performance assessment systematically analyzes what can happen, the likelihood, and the consequences. The U.S. Nuclear Regulatory Commission staff will use risk information to focus its review on those items most important to waste isolation. The staff will examine the U.S. Department of Energy identification of natural and engineered barriers important to waste isolation. The staff will use risk insights from previous performance assessments for the Yucca Mountain site, detailed process-level modeling efforts, laboratory and field experiments, and natural analog studies. The staff will then evaluate the U.S. Department of Energy scenario analysis. The scenario analysis must consider the risk information from identified barriers and include the identification and screening of features, events, and processes, and the construction of scenarios from the retained features, events, and processes of the Yucca Mountain site. Finally, the performance assessment review will examine information on 14 model abstractions. The abstractions arose from engineered, geosphere, and biosphere subsystems shown to be most important to waste isolation, based on previous performance assessments, knowledge of site characteristics, and repository design. The staff review will focus on those models and abstracts that are most risk significant to repository safety. For the postclosure period, “important to safety” means important to meeting the radiation exposure performance objective. The risk of radiation health effects is the basis for the radiation exposure limit. The postclosure performance objectives also protect ground water by limiting the radioactivity in a representative volume of ground water and require an assessment of performance under conditions of human intrusion.” (NRC 2003, page xvi)

4.3 References

CNWRA 2004. Mohanty, S., R. et. al. “System-Level Performance Assessment of the Proposed Repository at Yucca Mountain Using the TPA Version 4.1 Code.” CNWRA 2002–05. Rev. 2. Center for Nuclear Waste Regulatory Analyses (CNWRA), San Antonio, Texas. March 2004. ML041350316

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NRC 2001. “System-Level Repository Sensitivity Analyses Using TPA Version 3.2
Code.” NUREG 1746. U.S. Nuclear Regulatory Commission, Washington D.C. and Center for Nuclear Waste Regulatory Analyses, San Antonio, Texas. August 2001 ML012990126.

NRC (1995), ‘NRC Iterative Performance Assessment Phase 2 - Development of Capabilities for Review of a Performance Assessment for a High-Level Waste Repository’, *NUREG-1464*. ML040790450

NRC (1992), ‘Initial Demonstration of the NRC's Capability to Conduct a Performance Assessment for a High-Level Waste Repository’, *NUREG-1327*. ML012980272

5.0 DOE's License Application

DOE submitted its license application to the NRC seeking authorization to construct a geologic repository at Yucca Mountain on June 8, 2008 (DOE 2008ab). DOE provided the NRC its Safety Analysis Report (SAR) and additional information for the licensing proceeding that included the following key documents[3]:

1) General Information regarding the development of the potential repository at Yucca Mountain.

2) SAR Chapter 1: Repository Safety Before Permanent Closure

3) SAR Chapter 2: Repository Safety After Permanent Closure

4) SAR Chapter 3: Research and Development Program to Resolve Safety Questions

5) SAR Chapter 4: Performance Confirmation Program

6) SAR Chapter 5: Management Systems

7) SAR Appendix A: Information Designated as Official Use Only

8) Classified portion of the SAR was provided as a separate document that provided information related to naval nuclear fuel.

9) DOE's *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (FEIS) (dated February 2002) and stated it would submit a final supplement to the FEIS (including the Rail Alignment EIS), which was under development and was expected to be available on or before June 30, 2008.

10) The primary reference documents of the SAR (Enclosure 3 to the DOE submission includes a listing of the 196 reference documents

11) A matrix cross-referencing the SAR to 10 CFR Part 63 and the Yucca Mountain Review Plan (NUREG-1804)

DOE did provide an update to the license application (LA) on February 19, 2009 (DOE 2009av) and stated that "[C]hanges made to this revision were determined not to be significant and did not impact the conclusions of the LA."

The sections below provide on overview of DOE's Yucca Mountain license application that provides a discussion of: (i) the Yucca Mountain site (Section 5.1.1), (ii) the phased approach for repository development, should a construction authorization be granted, and the performance confirmation program the would be conducted throughout the development period (Section 5.1.2), (iii) safety during the operational or preclosure period (Section 5.1.3), (iv) activities associated with decommissioning of the surface facilities, permanent closure, and termination of the license (Section 5.1.4), (v) safety after permanent closure or the postclosure period (Section 5.1.5), and (vi) DOE updates of its license application (Section 5.1.6).

5.1 Overview of DOE's Yucca Mountain License Application

DOE's License Application (LA) represents a first of a kind type of facility for the geologic disposal of HLW. Key aspects of the license application are: (a) the Yucca Mountain site, (b) the extended time period for development of the repository (i.e., an estimated 100 years for construction, operation and permanent closure

[3] All the documents are available (except for the classified material) in the ADAMS Package# ML081560400.

of the repository), (c) the operational period that involves receipt and handling of HLW at surface facilities and emplacement of waste in the underground portion of the geologic repository operations area (GROA), (d) the very long time period for evaluating performance of the postclosure barriers (i.e., one-million year period after repository closure), and (e) the performance confirmation program period that continues to test and collect information for the critical evaluation of the safety basis (i.e., the confirmation program will continue over the 100 year operational period and supports the decision to permanently close the repository).

5.1.1 Yucca Mountain Site

Yucca Mountain is located in the southern Great Basin, within the Basin and Range Geological Province of the western United States. It is located on federal land in Nye County in southern Nevada, a semi-arid to arid region of the United States, approximately 90 mi northwest of Las Vegas. Figure 1 shows the footprint of the GROA with respect to the three federal properties that have a portion of the GROA within their boundaries, namely the Nevada Test Site (the Nevada Test Site was renamed as the Nevada National Security Site in 2010), Nevada Test and Training Range, and Bureau of Land Management property.

> "Yucca Mountain consists of a series of north–south-trending ridges extending approximately 25 mi from Timber Mountain in the north to the Amargosa Desert in the south. Above the location of the underground facility, the crest of Yucca Mountain reaches elevations of 4,920 to 6,330 ft above sea level. The west side of the crest is marked by an escarpment that drops approximately 1,000 ft to the base of Solitario Canyon. East of the crest, the mountain slopes gently to the east and is incised by a series of east- to southeast-trending washes. The elevation at the base of the eastern slope is approximately 1,100 to 1,500 ft below the ridge crest (Potter et al. 2002).
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> Yucca Mountain consists of successive layers of fine-grained volcanic rocks called tuffs, millions of years old, underlain by older carbonate rocks. These tuffs were formed when hot volcanic gas and ash erupted and flowed quickly over the landscape or settled from the atmosphere. In instances when the temperature was high enough, the ash was compressed and fused to produce a welded tuff. Nonwelded tuffs, which occur between welded layers, were compacted and consolidated at lower temperatures.
>
> The rock units at Yucca Mountain are classified according to geologic stratigraphy, hydrogeologic properties, and thermal-mechanical characteristics. Geologic stratigraphy is used for mapping; hydrogeologic properties are used for studies of water movement and potential radionuclide transport; and thermal-mechanical characteristics are used for evaluating rock strength, mechanical properties, and the effects of heat on mechanical properties.
>
> The repository horizon is located in the unsaturated zone a minimum of about 690 ft (210 m) above the water table in the present-day climate (SAR Section 1.1.4.2.3). It is expected to be more than 617 ft (188 m) above the water table during wetter future climate conditions (DOE 2008, SAR Section 2.3.5.2)
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> Understanding the meteorology and climatology at Yucca Mountain is necessary to evaluate the amount of water available to potentially interact with the waste. Both present-day and projected future climates are characterized by mean annual precipitation rates over the full infiltration modeling domain ranging from approximately 130 to 485 mm/yr. Evaporation, transpiration, and runoff reduce the amount of precipitation that infiltrates into the unsaturated zone typically by more than 90%. Mean annual surface infiltration rates range from less than 2 to about 83 mm/yr, with the largest values in projected cooler, wetter future climates (SNL 2008, Section 6.5)."   (DOE 2008 General Information, pages 1-2 to 1-3)

In short, the Yucca Mountain repository is located in a semi-arid to arid region of the United States and 100s of meters above the water table.

5.1.2 Phased Development and Performance Confirmation

NRC's regulations for geologic disposal at 10 CFR Part 63 "provides for a multi-staged licensing process that affords the Commission the flexibility to make decisions in a logical time sequence that accounts for DOE collecting and analyzing additional information over the construction and operational phases of the repository. The multi-staged approach comprises four major decisions by the Commission: (1) Construction authorization; (2) license to receive and emplace waste; (3) license amendment for permanent closure; and (4) termination of license. The time required to complete the stages of this process (e.g., 50 years for operations and 50 years for monitoring) is extensive and will allow for generation of additional information. Clearly, the knowledge available at the time of construction authorization will be less than at the subsequent stages. However, at each stage, DOE must provide sufficient information to support that stage." (66 FR November 2, 2001; 55738-39)

DOE provided its schedule for activities over the extended operational (or preclosure) period that included activities such as development of waste handling facilities, receipt and emplacement of waste, construction of the subsurface facility, monitoring, and closure activities (see Figure 2). Additionally, Figure 2 shows the performance confirmation continues throughout the operational period till the commencement of the permanent closure activities.

NRC stated

> "the general requirements at § 63.131(a) allow DOE the flexibility to develop and implement an effective performance confirmation program focused on confirming assumed subsurface conditions and assumed functionality of geologic and engineered systems and components important to postclosure performance (i.e., performance of barriers important to isolation) and/or preclosure repository operations (e.g., retrievability)." (66 FR November 2, 2001; 55744) DOE's SAR Chapter 4 (Performance Confirmation Program) "identifies 20 activities for performance confirmation. Some activities were selected as being most relevant to confirming preclosure and postclosure performance, based on current technical information and total system performance assessment results. Other activities were chosen to meet specific requirements described in 10 CFR 63, Subpart F. The decision analyses that resulted in selecting these activities will be periodically reassessed, based on updated technical information and total system performance assessment results, to assure the activities' continued relevance. New activities may be added, and currently planned activities may be curtailed or deleted as a result of these reassessments. The current conceptual descriptions of these activities will be supplemented by performance confirmation test plans that provide the rigor necessary to justify the activity, plan the details of its implementation, and establish condition limits for results that indicate significant differences from baseline information. Performance confirmation test plans have been written for seismic monitoring, precipitation monitoring, and construction effects monitoring. Other test plans will be prepared sequentially, and Performance Confirmation Plan (SNL 2008a) will be revised and updated as program development continues." (DOE 2008 SAR Chapter 4, page 4-3)

DOE's performance confirmation program includes: precipitation and seepage monitoring; subsurface water and rock testing; drift inspection; thermally accelerated drift near-field monitoring; seismicity monitoring; construction effects monitoring; waste package monitoring; corrosion testing; and waste form testing. This program will continue over the 100-year preclosure period to provide the Commission with updated and additional information for making the decision to permanently close the repository that DOE estimated at the time of submission of the license application in 2008 would occur in 2117.

5.1.3 Operational (or Preclosure) Safety

DOE presented descriptions for structure, systems, and components (SSCs), safety controls, and operational process activities for waste handling during the preclosure period involving: civil and structural systems; mechanical systems and electrical power systems; heating, ventilation, and air conditioning (HVAC) systems; instrumentation and control (I&C) systems radiation/radiological monitoring systems (RMS); and types of radioactive waste and the waste containers.

DOE explained:

> "[T]he GROA surface facilities have been designed to support a mostly canistered waste stream. A TAD canister is utilized for commercial SNF assemblies. The repository objective is to have 90% of individual commercial SNF assemblies loaded into TAD canisters by the utilities with a limited quantity of uncanistered individual commercial SNF assemblies and dual-purpose canisters requiring handling in a pool (i.e., submerged). In some cases, commercial SNF will require aging before it is ready for emplacement.
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> Canistered waste (HLW and naval SNF) is delivered to the Initial Handling Facility in transportation casks. The naval SNF canisters will be delivered via railcar; HLW will be delivered by railcar or legal-weight trucks. Equipment in the Initial Handling Facility unloads the cask from the conveyance, removes the canister from the cask, and places the empty cask back onto the conveyance. The canister is loaded into a waste package, which is welded closed. The sealed waste package is transferred to the TEV, which transports the waste package to the emplacement drifts within the mountain.
>
> The Wet Handling Facility will process the limited number of uncanistered commercial SNF assemblies received from utilities and DOE sites. The uncanistered assemblies will be received in dual-purpose canisters and transportation casks. The dual-purpose canisters will be opened and the commercial SNF inside will be transferred under water to TAD canisters or staging racks in the pool. Uncanistered commercial SNF shipped in casks will also be transferred underwater to TAD canisters or staging racks. Once the assemblies are loaded into the TAD canister, the TAD canister, in a shielded transfer cask, will be removed from the pool and transferred to the TAD closure station. Once in the TAD closure station, a portion of the water in the TAD canister is removed to allow welding of the inner lid. Once the inner lid is sealed, the remaining water internal to the TAD canister is displaced with helium and the internal volume of the TAD canister is dried. Once dried, the canister is backfilled with helium, the vent and drain connections are sealed, and the outer lid is welded on. Upon completion of these steps, the closed TAD canister is transferred, by a canister transfer machine, from the shielded transfer cask to an aging overpack for transfer to a Canister Receipt and Closure Facility for loading into a waste package for disposal or for transfer to an aging pad." (DOE 2008 General Information, page 1-6)

Figure 3 presents the different types of facilities that would comprise the GROA facility. Figure 3 also depicts an initial operating capability and full operating capability – the objective of the initial phase is to is to develop the capability to start operations, including development of those assets necessary to achieve a reasonable ramp-up of operations during the first several years of waste receipt, whereas the objective of the subsequent phases is to develop full operating capability for receiving and emplacing the 70,000 MTHM currently authorized by law for the repository. A phased approach provides opportunities for implementing lessons learned for use on subsequently constructed facilities.

The handling of nuclear fuel assemblies and canister fuel is not unique to a repository. DOE explained how the use of industry experience and precedent were utilized in providing for safe operations that was quantified in its Preclosure Safety Analysis (PCSA).

> "Preclosure safety is ensured by the application of numerous safety principles that contribute to the protection of public health and safety, the environment, and worker safety. The attributes of the repository site combine with the design of the repository structures, systems, and components (DOE presented descriptions for SSCs, safety controls, and operational process activities for waste handling during the preclosure period involving: civil and structural systems; mechanical systems and electrical power systems; heating, ventilation, and air conditioning (HVAC) systems; instrumentation and control (I&C) systems radiation/radiological monitoring systems (RMS); and types of radioactive waste and the waste containers. s) to achieve safety by maximizing the prevention of events and minimizing the reliance on immediate automatic or human actions.
>
> The Yucca Mountain repository site is located on federal land with the site boundary approximately 5 mi away from the waste handling facilities and currently with no permanent residents within approximately 14 mi of the geologic repository operations area. This remoteness from the general public reduces the potential effects of the events considered in the safety analysis. Even though the site is remote, the prevention and mitigation features of the design and operation are consistent with those of facilities with more radioactive material at risk and closer proximity to the general public.
>
> To the extent practicable, the repository design is based on proven nuclear industry precedent and utilizes primarily canistered spent nuclear fuel (SNF) to minimize handling of individual fuel assemblies. Facility components are designed with robust margins and utilize diverse and redundant systems. Mechanical handling, shielding, and related safety equipment are based on proven technology. The safety philosophy includes design approaches where (1) prevention is preferable to mitigation, (2) design features are preferable to administrative features, (3) passive features are preferable to active features, and (4) automatic features are preferable to manual features. SSCs that are important to safety (ITS) are designed with sufficient margin and reliability that an event sequence resulting in the exposure of workers or the public to radiation is maintained at a low probability.
>
> The PCSA provides a framework for risk-informed, performance-based decision making that is applied to identify SSCs that are ITS; to identify measures for providing defense in depth; and to identify license specifications to ensure operation consistent with the SAR. The PCSA identifies the potential natural and operational hazards for the preclosure period; assesses potential initiating events and event sequences and their consequences; and identifies the SSCs and procedural safety controls intended to prevent or reduce the probability of an event sequence or mitigate the consequences of an event sequence, should it occur. Specific design features that perform these functions are identified. The design information and analyses must be sufficient to demonstrate that the design features will perform their intended safety functions. Initiating event and event sequence identification and analysis comprise an iterative process integrally tied to repository design. The results of the PCSA (see Sections 1.6 to 1.9) confirm that the site characteristics combined with the repository design provide an inherently safe facility that meets the regulatory preclosure performance objectives with substantial margin." (DOE 2008 SAR Chapter 1: Repository Safety Before Permanent Closure; page 1-1)

5.1.3.1 Underground Facility and Operations

DOE's schedule for construction of the underground facility (e.g., construction of drifts where waste is to be emplaced) requires the concurrent performance of construction and repository nuclear operations (i.e., emplacement of waste). DOE stated:

> "To ensure the safety of project personnel and operational security, it will be necessary to separate these activities. During the construction process, separation is maintained by designing independent systems for repository operations and construction. This includes sufficient physical space between construction and operation activities to prevent impact. Protected area boundaries with physical barriers and isolation zones isolate personnel movement between nuclear operations and construction areas.
>
> The emplacement operations take place in finished emplacement drifts at the same time that emplacement drifts are being constructed in other underground areas behind physical barriers. During construction, isolation bulkheads and separate ventilation systems between the development side (i.e., drifts under construction) and the emplacement side (i.e., the drifts where waste packages are being emplaced) minimize the risk of worker exposure to radiation from the waste and ensure the protection of emplaced waste packages from construction hazards. Air pressure on the development side is maintained higher than the pressure on the emplacement side in order to prevent potential airborne radioactive contamination movement from the emplacement side to the development side. Construction personnel and material do not enter the underground facility through portals and mains used to move waste packages underground. Drift construction is supported by facilities outside of the GROA, such as the South Portal development area and the North Construction Portal area." (DOE 2008 General Information, page 1-17)

Another aspect of the underground facility is the relationship between the surface aging pad located within the GROA and emplacement operations. An aging facility was considered necessary to allow DOE flexibility in control the heat load in individual drifts. In particular, because there is uncertainty about the thermal output of individual packages arriving at the repository over time, availability of an aging pad provides flexibility in managing waste emplacement to achieve a uniform heat loading in a drift consistent with DOE's design objectives. DOE explained:

> "The purpose of the aging facilities is to provide safe cooling of commercial SNF within TAD canisters and dual-purpose canisters, in aging overpacks or horizontal aging modules, until the thermal heat load of the SNF has decayed to a level low enough to be placed in a waste package. The aging facility includes aging overpacks and crawler-type site transporters for moving aging overpacks containing canisters of commercial SNF between aging pads and various surface facilities" (DOE 2008 General Information, page 1-5)

One aspect of waste emplacement that is different from other handling operations is the stability of the waste emplacement drifts. DOE evaluated the stability of underground emplacement areas, including mechanical and thermo-mechanical effects. DOE concluded and NRC staff concurred in their preclosure SER that the emplacement areas would remain stable during the preclosure period.

5.1.3.2 Retrieval Plans

> "The GROA is designed to permit retrieval of any or all emplaced waste, starting at any time up to the beginning of permanent closure. Retrieval operations could result from a demonstration by the Performance Confirmation Program that the postclosure regulatory standard may not be met or a policy decision to recover the economically valuable contents of SNF or to dispose of waste in a different manner. For planning purposes, it is assumed that closure and decommissioning activities begin approximately 10 years prior to closure.

> The Performance Confirmation Program monitors subsurface conditions and performs tests to confirm geotechnical and design assumptions to provide information to allow actions to preserve the retrievability option. The design approach to satisfy this requirement is to ensure the repository design and emplacement operations do not preclude the retrieval of any or all waste packages prior to closure of the repository. Retrieval, as defined in 10 CFR 63.2, is 'the act of permanently removing radioactive waste from the underground location at which the waste had been previously emplaced for disposal.'
>
> If a retrieval decision is made, waste would be placed in a storage or disposal facility designed in accordance with the regulations that are applicable at the time. GROA aging pads with space for up to 2,500 waste packages, SNF or HLW, are available for retrieved material but would require a specially designed waste package overpack. Ample additional space within or near the GROA is available to develop waste package storage, as needed." (DOE 2008 General Information, page 1-18)

5.1.4 Decommissioning, Permanent Closure, and License Termination

After repository operations and the Performance Confirmation Program have been completed, the DOE would need to submit an application with the NRC for a license amendment to close the repository in accordance with 10 CFR 63.51 (License Amendment for Permanent Closure). The license amendment will address the activities associated with the decommissioning and permanent closure. DOE stated that it would determine at that time that the surface facilities are no longer required to support SNF and HLW handling, processing, emplacement, or retrieval and could be decommissioned and "upon NRC approval of the application, the surface facilities, except for permanent monuments and markers, will be decontaminated, decommissioned, and dismantled" (DOE 2008 General Information, page 1-19).

In addition to decommissioning, activities to permanently close the repository would address requirements in Part 63. In particular 10 CFR 63.51(a)(2) and (3) require that the DOE undertake measures to regulate or prevent activities that could impair long-term waste isolation and that the repository institute a monitoring program after permanent closure. DOE explained that a network of permanent monuments and markers will be erected in various areas at the site to warn future generations of the presence and nature of the buried waste, and detailed public records will identify the location and layout of the repository and the hazardous nature of the waste it contains (SAR Section 5.8). For planning purposes DOE assumed that closure and decommissioning activities would begin approximately 10 years prior to closure (DOE 2008 General Information, page 1-18).

Following permanent closure and the decontamination or decontamination and dismantlement of surface facilities at the Yucca Mountain site, DOE may apply for an amendment to terminate the license consistent with 10 CFR Part 63.52. The Commission explained license termination when it finalized its 10 CFR Part 63:

> "License termination represents the end of NRC involvement with the repository. However, the Commission would not terminate the license unless and until all requirements have been met by DOE. License termination removes NRC oversight of the Yucca Mountain site, leaving DOE as the single Federal authority responsible for the site. Under the proposed part 63, the license amendment for permanent closure must include a DOE program for continued oversight to prevent any activity at the site that poses an unreasonable risk of breaching the geologic repository's engineered barriers or increasing radiation exposure of individual members of the public beyond allowable limits." (66 FR 55739; November 2, 2001)

DOE provided its plans for continued oversight and monitoring during the postclosure period, including preservation of records and permanent markers in SAR Section 5.8. DOE provided its initial plans; however, detailed final designs and plans are not needed at this time recognizing closure activities are anticipated to be needed approximately 100 years into the future and it would be expected that materials, designs, and approaches would be finalized consistent with technological capabilities at that time.

5.1.5 Postclosure Safety

DOE described its multiple barrier approach for demonstrating repository safety over the postclosure period, that is the time after the repository is permanently closed and reliance for safety is based on the repository's passive safety barriers.

> "The performance of a repository at Yucca Mountain is controlled by the natural and engineered features of the site that act in concert to prevent or reduce the movement of water and/or the transport of radioactive materials to the accessible environment. Multiple natural features of the Yucca Mountain site and engineered features of the repository design combine to form the following three barriers important to waste isolation: the Upper Natural Barrier, the EBS, and the Lower Natural Barrier. The Upper Natural Barrier includes the geologic units from the surface to the repository horizon, including alluvial soils and gravel, the Tiva Canyon welded tuff, the Paintbrush nonwelded tuff, and the Topopah Spring welded tuff. The EBS is composed of the manmade features within the emplacement drifts, including the drip shield, waste package, waste form, and other engineered components. The Lower Natural Barrier includes the unsaturated and saturated volcanic tuff units below the repository and older bedrock units and alluvial deposits below the water table between Yucca Mountain and the accessible environment in Amargosa Valley.
>
> The geologic and hydrologic characteristics of the Yucca Mountain site form effective natural barriers to the flow of water and to the potential movement of radionuclides. The underground environment within the natural setting is conducive to the design and construction of components that prevent or reduce the movement of water or the potential release and transport of radionuclides. The waste isolation capability of the natural setting at the site is a direct function of the favorable intrinsic characteristics of the geologic units and their durability. The capability of the EBS is achieved by designing components specifically to function in the natural setting of the Yucca Mountain repository, particularly its unsaturated rock units. The materials in the EBS have been chosen so that the components perform their intended functions for many thousands of years. The barriers in the repository system work individually and together to prevent or substantially reduce the rate of movement of water, the release rate of radionuclides from the waste, and the rate of movement of radionuclides from the repository to the accessible environment. Analyses of both the natural barriers and the EBS address their effectiveness both during the first 10,000 years after closure and for the period beyond 10,000 years, within the period of geologic stability as prescribed by proposed 10 CFR Part 63." (DOE 2008, SAR Chapter 2: Repository Safety after Permanent Closure; page 2-2)

Figure 4 illustrates the multiple barriers in DOE's safety analysis report for Yucca Mountain.

5.1.5.1 Performance Assessment

DOE evaluated the performance of the repository after permanent closure with its Total System Performance Assessment (TSPA) that, by regulation (10 CFR Part 63.114) is required to include the relevant features, events and processes (FEPs) affecting repository performance subject to certain limitations at 10 CFR Part 63.342 [e.g., DOE shall not include consideration of very unlikely features, events, or processes, i.e., those that are estimated to have less than one chance in 100,000,000 per year of occurring]. DOE stated:

> "The TSPA model incorporates and integrates models describing the characteristics, features and processes associated with the three barriers (Upper Natural Barrier, EBS, and Lower Natural Barrier). Sections 2.1.1 and 2.1.2 contain a description of these three barriers, and a summary of the associated features and processes. Section 2.3 provides a much more in-depth description of the various physical phenomena, thermal-hydrologic-chemical-mechanical couplings, and modeling abstractions for these features and processes (as well as the likely and unlikely disruptive events associated with the Yucca

Mountain site). The TSPA approach combines these underlying abstractions in such a way that it incorporates the estimated ranges of uncertainty in the parameter distributions, model abstractions, and disruptive events and then propagates this uncertainty into estimates of the annual dose.

The TSPA model was built expressly to evaluate the Yucca Mountain repository system in accordance with the requirements of proposed 10 CFR Part 63. The first step in building the model, consistent with the definition of performance assessment in proposed 10 CFR 63.2 and requirements in proposed 10 CFR 63.114(a)(4) to (6), is to identify the FEPs that could be important to repository performance. As specified in the proposed 10 CFR 63.342, the performance assessments for the human intrusion and groundwater protection standards do not include consideration of unlikely FEPs (those with a greater than one chance in 10,000 of occurring in 10,000 years but less than a one chance in 10 of occurring in 10,000 years). However, the performance assessment for the individual protection standard includes both likely and unlikely FEPs, and only excludes very unlikely FEPs (those with less than one chance in 10,000 of occurring in 10,000 years) or those with low consequence (proposed 10 CFR 63.114(a)(4) to (6)). Furthermore, the TSPA model and associated performance assessment described in this section expressly follow the requirements in proposed 10 CFR 63.342(c) by projecting the continued effects of the 10,000-year screened-in FEPs through the period of geologic stability (up to 1,000,000 years after permanent closure), and including the effects of seismic events, igneous events, climate change, and general corrosion beyond 10,000 years.

The TSPA is built upon FEPs that have been identified and screened in accordance with the requirements in proposed 10 CFR 63.114(a)(4) to (6). FEPs are included or excluded based upon the three screening criteria described in Section 2.2.1.2: low probability (proposed 10 CFR 63.342(a)), low consequence (proposed 10 CFR 63.114(a)(5)), and regulation. Each screening decision is supported by a technically sound screening justification, as described in detail in *Features, Events, and Processes for the Total System Performance Assessment: Methods* (SNL 2008b) and *Features, Events, and Processes for the Total System Performance Assessment: Analyses* (SNL 2008c). The FEPs screening methodology is summarized in Section 2.2.1, and the FEPs screening decisions (i.e., inclusion or exclusion), along with a brief description of each FEP, can be found in Table 2.2-5. Summaries of the technical basis and justification for each included FEP screening decision can be found in the FEPs inclusion tables of each Section 2.3 subsection (e.g., Table 2.3.1-1) and in Section 2.2 (i.e., Table 2.2-4) for 'system' FEPs.

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[t]he TSPA calculates the total annual dose as the sum of the annual doses attributable to the nominal scenario class, the early failure scenario class, and the two disruptive event scenario classes (the igneous scenario class and the seismic scenario class). Computation of the dose attributable to each scenario class relies on the separation of each disruptive-event scenario class (as well as the early failure scenario class) into two modeling cases, each of which is built around a more narrowly defined event occurrence. For example, the volcanic eruption modeling case calculates the contribution to the total annual dose from the set of futures within the broader igneous scenario class that have one or more atmospheric eruptions occurring in them. The six modeling cases associated with the aforementioned event scenario classes are as follows: igneous intrusion, volcanic eruption, seismic ground motion, seismic fault displacement, early-failure of waste packages, and early-failure of drip shields. In addition, a seventh modeling case describes performance in the absence of disruptive or early failure events, and is called the nominal modeling case. (DOE 2008 SAR Chapter 2 pages 2.4-10 and 11)

Figures 5 through 9 provide a pictorial representation of the modeling approach in the TSPA and for specific scenarios. Figure 10 shows the contribution of each of the modeling cases to the total mean annual dose and shows the estimated doses are well below the individual protection standards for the initial 10,000 years (i.e., 15 mrem/yr) and for the period after 10,000 years (i.e., 100 mrem/yr). In particular, the peak mean dose[4] was estimated to be approximately 0.2 mrem/yr over the initial 10,000 years and 2 mrem/yr over the period after 10,000 years. The dose estimates included the consideration of uncertainty. [Note: the nominal scenario is not depicted in Figure 10 because the nominal scenario class does not contribute to dose for the initial 10,000 years (SAR Page 2.4-62) and the nominal scenario is combined with the seismic scenario class due to the high probability of this scenario class for the period after 10,000 years (SAR page 2.4-36)] DOE explained in its SAR:

> "The TSPA separates quantitative uncertainty in model inputs into two categories: aleatory uncertainty and epistemic uncertainty. Aleatory uncertainty primarily refers to the inherent uncertainty regarding the timing and magnitude of future events that could affect the repository and the impact of these events on repository performance. Because aleatory uncertainty cannot be reduced by the acquisition of additional data or knowledge, this kind of uncertainty is also referred to as irreducible uncertainty. Examples of aleatory uncertainty considered in the TSPA include the time and amplitude of seismic ground motion events, the occurrence of igneous events, and the location and number of early failures of waste packages and drip shields due to undetected manufacturing or emplacement defects.
>
> The other important type of uncertainty is called epistemic uncertainty and stems from a lack of knowledge about a parameter or a probability distribution that is believed to be fixed (or deterministic). Sources of epistemic uncertainties include incomplete data, estimates based upon expert judgment, and measurement errors. Unlike aleatory uncertainty, epistemic uncertainty is potentially reducible with additional data and knowledge. In the TSPA model, epistemic quantities are generally inputs to specific submodels, with the submodels having been developed to use single values for these quantities. A particular epistemic quantity can be a parameter that characterizes a probability distribution (e.g., the mean value of the fracture permeability distribution used to calculate drift seepage), a field of values selected from alternative sets (e.g., the flow field in the unsaturated zone), or a measured parameter that characterizes a physical-chemical process (e.g., the temperature dependency of general corrosion of Alloy 22 (UNS N06022) or the unsaturated-zone fracture frequency)." (DOE 2008 SAR Chapter 2 page 2.4-6)

Figure 11 provides a depiction of the incorporation of aleatory and epistemic uncertainty in the seismic modeling case. DOE explained that a variety of the statistical measures can be derived from its uncertainty approach, including the mean, median, and 5th and 95th percentile curves. Such an approach was considered useful, in part, due to the proposed EPA Standard for the period after 10,000 years and NRC's proposed implementing regulations specified a median dose limit. In describing Figure 2.4-8 (reproduced as Figure 11 in this report), DOE explained:

> "The mean annual dose curve or history is plotted as a red curve and computed by taking the arithmetic average or expectation of the 300 expected annual dose values at each time T along the curves. Similarly, the median dose curve, plotted as a blue curve, is constructed by sorting the 300 expected values from lowest to highest at each time τ, and then averaging the two middle values. Curves for the 5th and 95th percentiles are also plotted to illustrate the uncertainty in the expected annual dose histories; 90% (or 270 of the 300 epistemic realizations) of the projected dose histories fall between these two percentile curves. For the first 10,000-year period after closure of the repository, as required by proposed 10 CFR 63.303 and 63.311, the actual "annual dose curve" referred to in Section 2.2.1.4 of NUREG-1804 is calculated to be the aforementioned arithmetic mean annual dose curve, while for post-10,000-year compliance, the median annual dose curve is calculated to determine compliance with the individual protection and human intrusion standards. The actual single value compliance metric in proposed 10 CFR 63.303 and proposed 10 CFR 63.311 (either 15 mrem/yr for 10,000-year compliance or 350 mrem/yr for post-10,000-year compliance) is

[4] "peak mean dose" is the maximum of the mean dose curve over time - the mean dose curve is the representation of the mean dose over time. In the probabilistic analysis of the repository each possible future consequence of the repository system is represented by a curve describing the annual dose as a function of time – the mean dose curve is developed by averaging the dose from each of the possible scenarios at a particular point in time for all times of interest.

either the maximum of the mean curve before 10,000 years or the maximum of the median curve after 10,000 years." (DOE 2008 SAR Chapter 2 pages 2.4-20 and 2.4-21)

5.1.5.2 Performance Assessment for Human Intrusion

The TSPA is also used to estimate the potential dose for demonstrating compliance with the regulatory requirements for human intrusion. The requirements at 10 CFR Part 63.321 specify that DOE must (1) determine the earliest time after disposal that the waste package would degrade sufficiently that a human intrusion could occur without recognition by the drillers and (2) demonstrate that there is a reasonable expectation that the reasonably maximally exposed individual receives, as a result of the human intrusion, no more than the an annual dose of 0.15 mSv (15 mrem) for 10,000 years following disposal; and 1.0 mSv (100 mrem) after 10,000 years, but within the period of geologic stability (i.e., 1-million years). The performance assessment for the human intrusion scenario is subject to certain requirements such as "intruders drill a borehole directly through a degraded waste package" (§ 63.322 - Human Intrusion Scenario) and this performance assessment excludes unlikely features, events, and processes or sequences of events and processes, i.e., those that are estimated to have less than one chance in 100,000 per year of occurring (see § 63.342 Limits on Performance Assessment).

DOE estimated the time for a human intrusion event to occur without recognition by the drillers would be in the very distant future:

> "Thus, for the TSPA human intrusion scenario, the human intrusion event time based on drip shield general corrosion and observable effects on the drilling system is considered to occur no earlier than 200,000 years. This is a conservative assumption, as there is only a 0.0001 chance of drip shield general corrosion failure prior to 230,000 years and because waste package degradation (formation of open breaches) takes significantly longer than this. Thus, until some time after 230,000 years, there is a double barrier to drilling penetration of the waste forms. For the stylized human intrusion scenario, 200,000 years is conservatively assumed to be the earliest time a driller could penetrate a waste package without recognition. The dose results described below use this conservative intrusion time, without uncertainty." (DOE 2008 SAR Chapter 2 pages 2.4-308 and 309)

DOE estimated the mean annual dose from the human intrusion to be approximately 0.01 mrem (see Figure 12).

5.1.5.3 Performance Assessment for Separate Standards for Ground-water Protection

The TSPA is also used to estimate the potential groundwater concentrations and doses for demonstrating compliance with the regulatory requirements for the separate standards for groundwater protection. The separate standards provide for dose limits for some radionuclides and concentration limits for other radionuclides. In particular, 10 CFR 63.331 includes groundwater protection limits for radionuclides in the representative volume of groundwater for 10,000 years after disposal for the following quantities:

- Combined Ra^{226} and Ra^{228} concentration, including natural background
- Gross alpha activity concentration (including 226Ra but excluding radon and uranium), including natural background
- Annual beta-photon dose to the whole body or any organ from drinking 2 liters of water per day, excluding natural background.

As with the human intrusion scenario the performance assessment for groundwater protection excludes unlikely features, events, and processes or sequences of events and processes, i.e., those that are estimated to have less than one chance in 100,000 per year of occurring (see § 63.342 Limits on Performance Assessment). Additionally, the representative volume of groundwater is specified to contain 3,000 acre-feet of water (about 3,700,000,000 liters or 980,000,000 gallons) at § 63.332 – Representative Volume. DOE estimated that releases from the repository were well below the three separate standards (i.e., combined Ra^{226} and Ra^{228}, gross alpha concentration, and annual beta-photon dose) for groundwater protection – see Figures 13-15.

5.1.6 License Application and its Updates

DOE submitted its license application on June 8, 2008 and identified 196 key documents supporting the license application (DOE 2008). DOE provided an update to the license application (LA) on February 19, 2009 and stated that "[C]hanges made to this revision were determined not to be significant and did not impact the conclusions of the LA" (DOE 2009).

DOE submitted the license application prior to the finalization of EPA's final standards and NRC final regulations for the period after 10,000 years. DOE did not supplement the application after standards and regulations were finalized for this time period, however, DOE responded to NRC's requests for additional information (RAIs) to address discrepancies between the final rule and the proposed rule for the period after 10,000 years. For example, NRC requested that DOE address differences associated with differences between the final and proposed rule related to the dose limit, the range for deep percolation rate, and water table rise due to seismic activity (DOE 2009cb Enclosure 6). As appropriate, these type of RAIs are identified in the Sections on NRC's safety evaluation report (SER).

5.2 References

DOE 2009cb. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 5." Letter (June 5) J.R. Williams to J.H. Sulima (NRC). Washington, DC: DOE, Office of Technical Management. ML091590581.

DOE 2009av. DOE/RW-0573 "Update to the Yucca Mountain Repository License Application (LA) for Construction Authorization." Rev. 1. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. ML090700817

.

DOE 2008ab. DOE/RW-0573 "Yucca Mountain Repository License Application." Rev. 0 Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. ML081560400

66 FR 55732 - 55816; November 2, 2001; 10 CFR Part 63 – Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada; Final Rule; U.S. Nuclear Regulatory Commission; Washington, D.C.

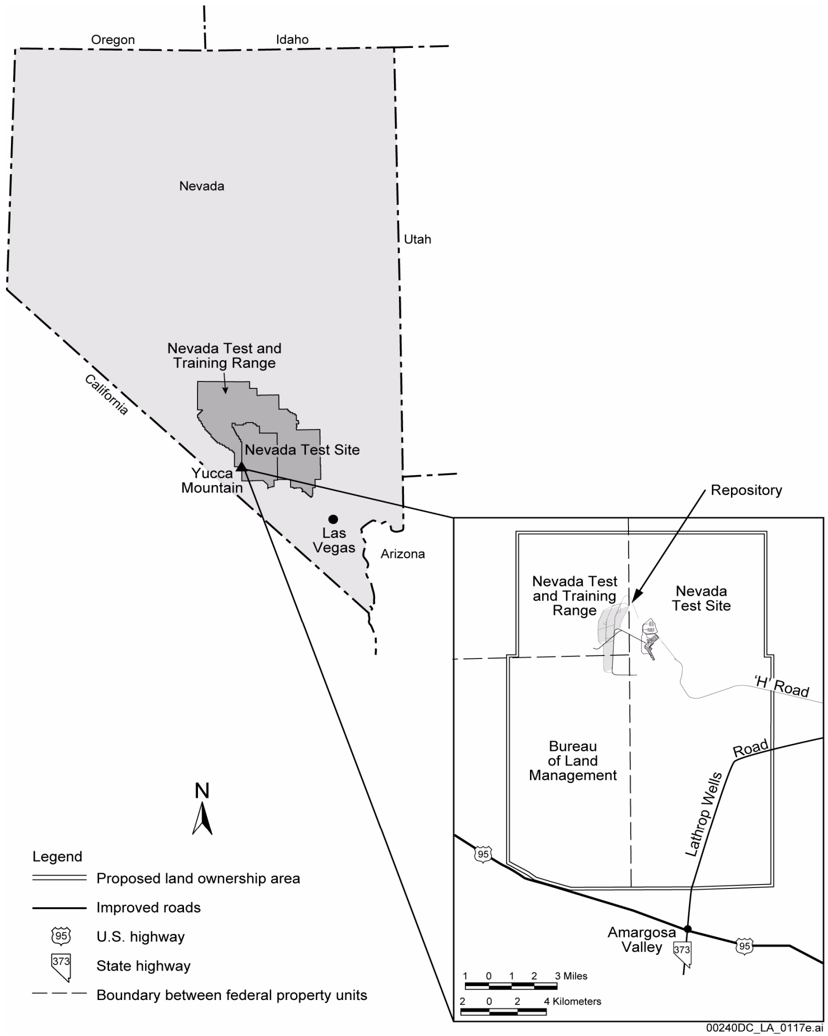
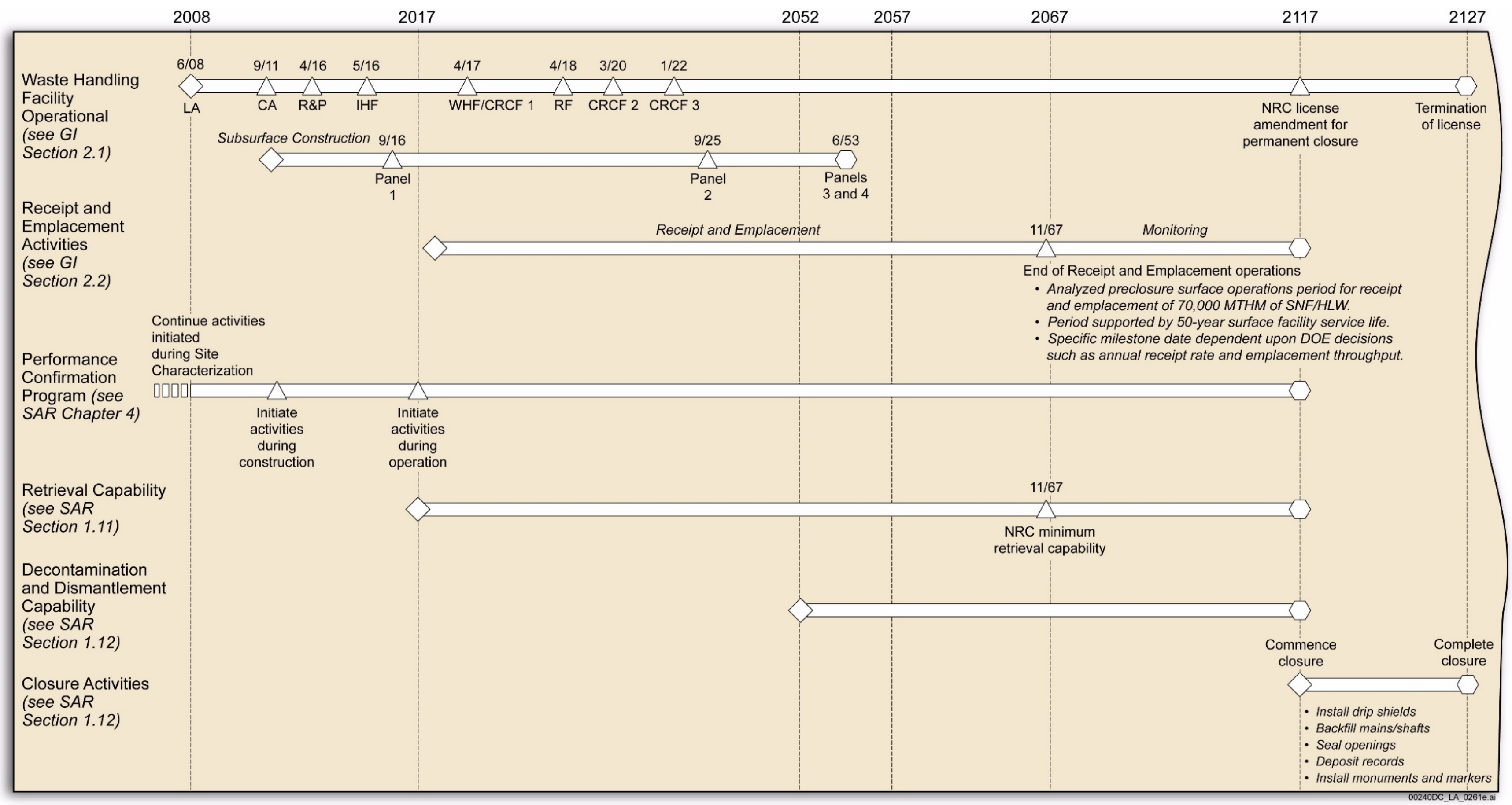


Figure 1. Federal land immediately surrounding Yucca Mountain (taken from General Information Document; DOE, 2008 Figure 1-2)



NOTE: CA = construction authorization; CRCF = Canister Receipt and Closure Facility; IHF= Initial Handling Facility; LA = license application; R&P = receive and possess; RF = Receipt Facility; WHF = Wet Handling Facility.

Figure 2. Repository Operations Summary Timeline (taken from General Information Document; DOE, 2008 Figure 1-7)

LEGEND			
Initial Operating Capability			
Phase 1			
050	Wet Handling Facility	26D	Emergency Diesel Generator Facility
060	Canister Receipt and Closure Facility 1	27A	Switchyard (138kV)
51A	Initial Handling Facility	27B	13.8kV Switchgear Facility
17R	Aging Pad R	28A	Fire Water Facility
160	Low-Level Waste Facility	28B	Fire Water Facility
220	Heavy Equipment Maintenance Facility	30A	Central Security Station
230	Warehouse and Non-Nuclear Receipt Facility	30B	Cask Receipt Security Station
240	Central Control Center Facility	33A	Rail Car Buffer Area
25A	Utility Facility	33B	Truck Buffer Area
25B	Cooling Tower	35A	Septic Tank and Leach Field
25C	Evaporation Pond	66A	Helicopter Pad
26B	Standby Diesel Generator Facility	290	Aging Overpack Staging Facility
90A	Storm Water Retention Pond		
Full Operating Capability			
Phase 2			
200	Receipt Facility	68B	Materials/Yard Storage
28E	Fire Water Facility	690	Vehicle Maintenance and Motor Pool
620	Administration Facility	70A	Diesel Fuel Oil Storage
63A	Fire, Rescue and Medical Facility	70B	Fueling Stations
65A	Administration Security Station	71A	Craft Shops
65B	Administration Security Station	71B	Equipment/Yard Storage
68A	Warehouse/Central Receiving		
Phase 3			
070	Canister Receipt and Closure Facility 2	17P	Aging Pad P
Phase 4			
080	Canister Receipt and Closure Facility 3	30C	North Perimeter Security Station

00240DC_LA_0233o.ai

Figure 3. Phased construction of surface facilities (DOE 2008 General Information, Figure 2-2)

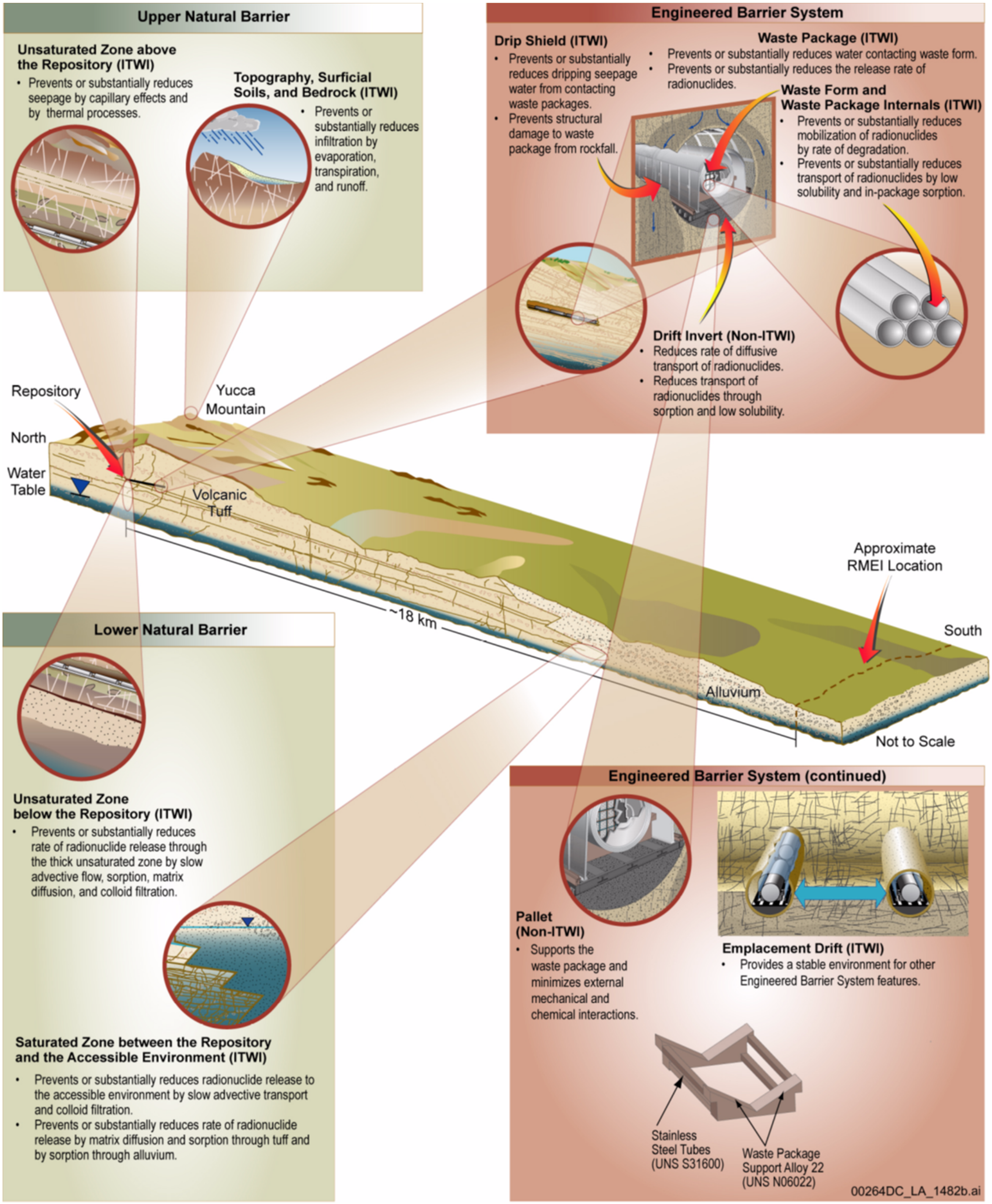


Figure 4. Illustration of the multiple barrier repository system (DOE 2008, SAR Chapter 2; Figure 2.1-1)

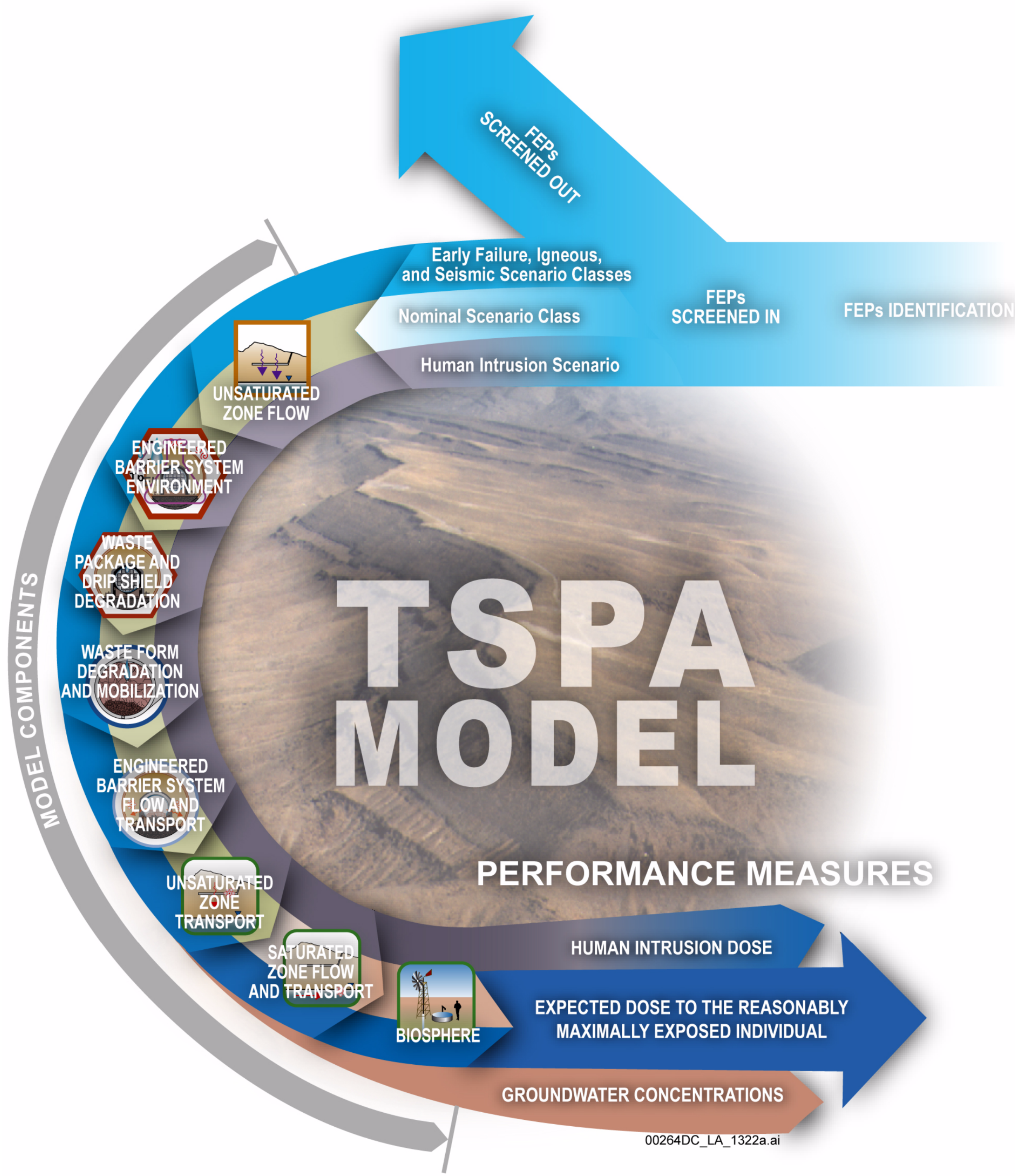


Figure 5. Schematic Representation of the Development of the TSPA Model, Including the Nominal, Early Failure, Igneous, and Seismic Scenario Classes, as Well as the Human Intrusion Scenario (DOE 2008, SAR Chapter 2; Figure 2.4-1).

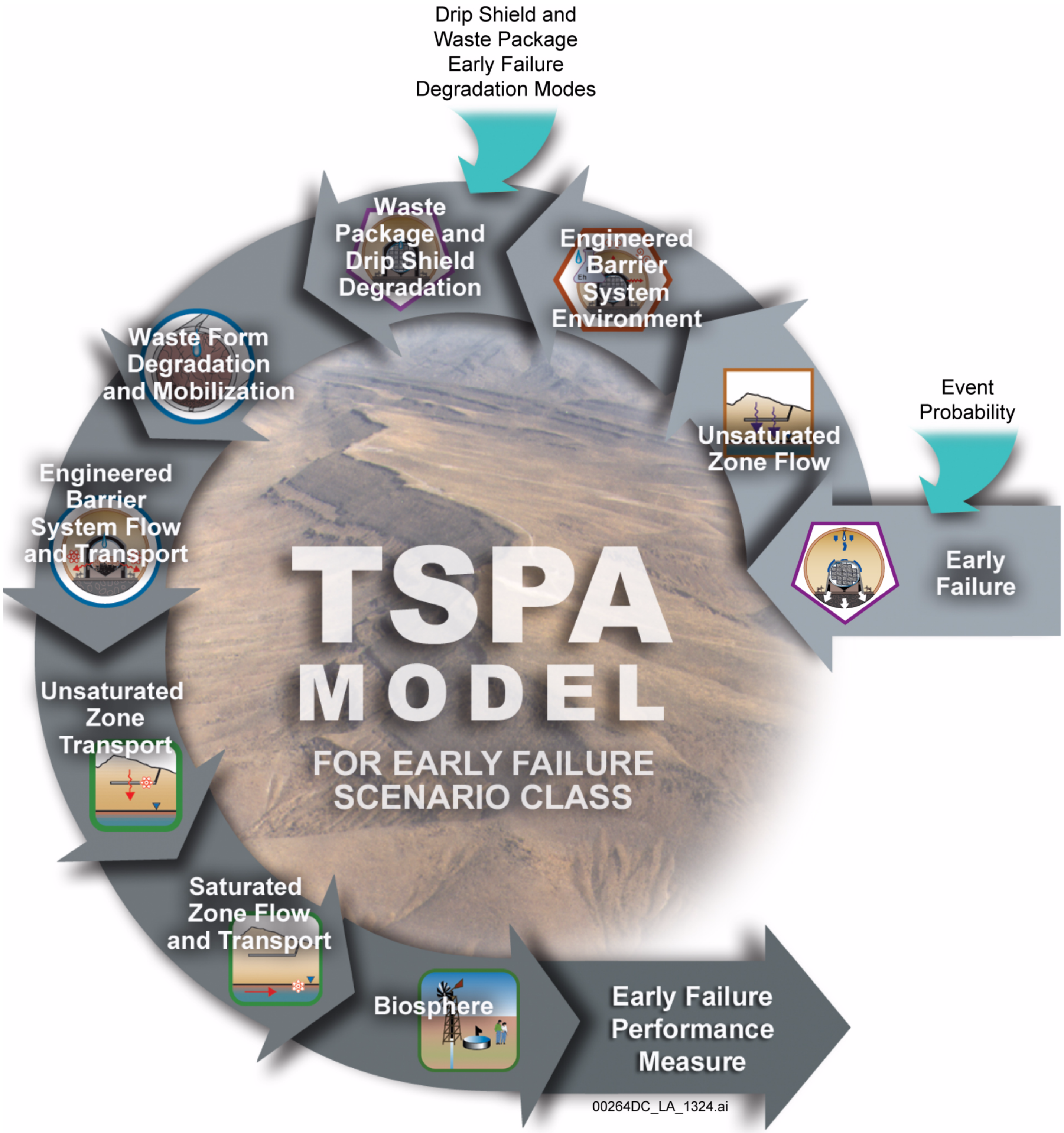


Figure 6. TSPA Model Components for the Early Failure Scenario Case (DOE 2008, SAR Chapter 2; Figure 2.4-4).

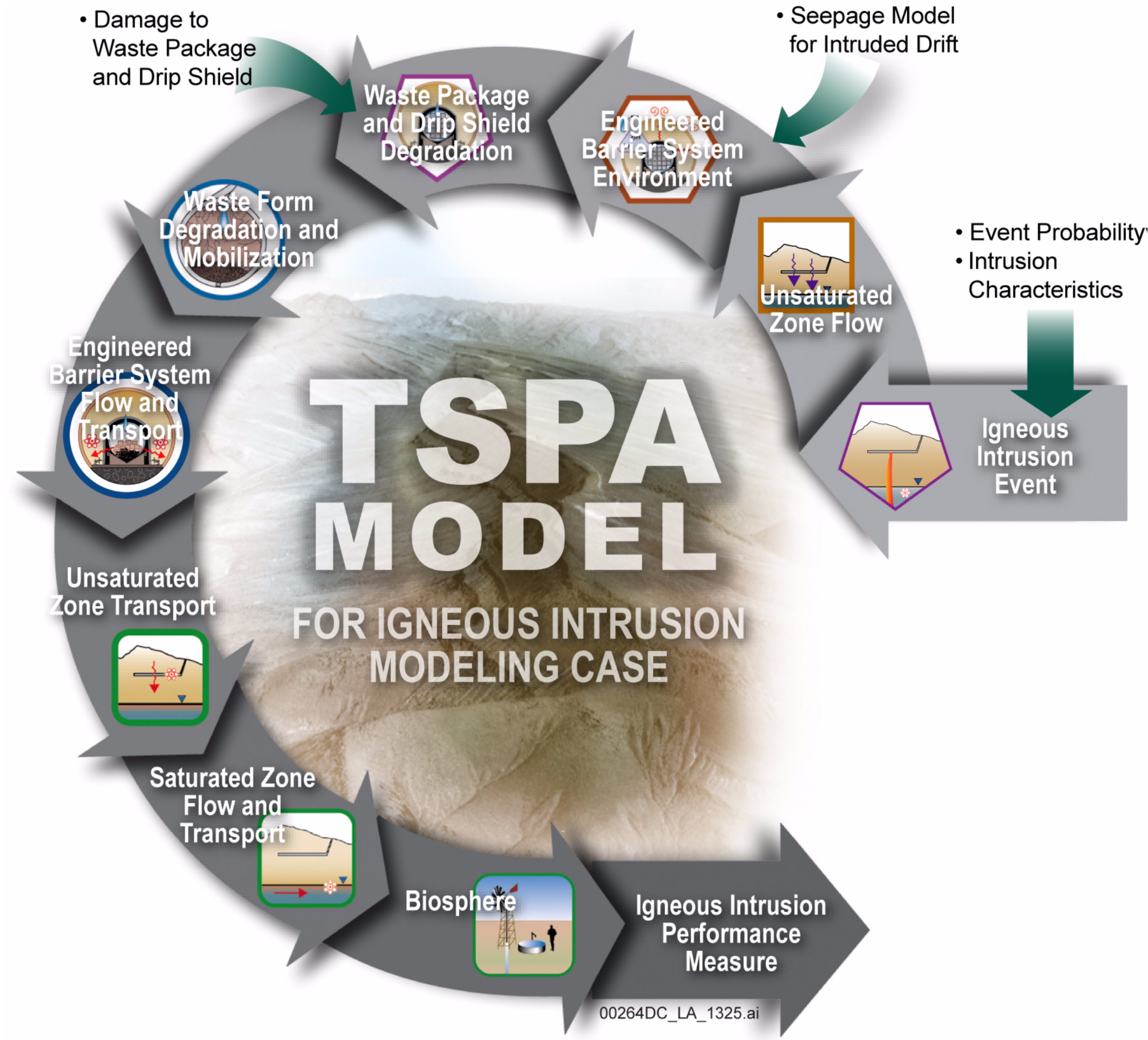


Figure 7. TSPA Model Components for the Igneous Intrusion Modeling case (DOE 2008, SAR Chapter 2; Figure 2.4-5).

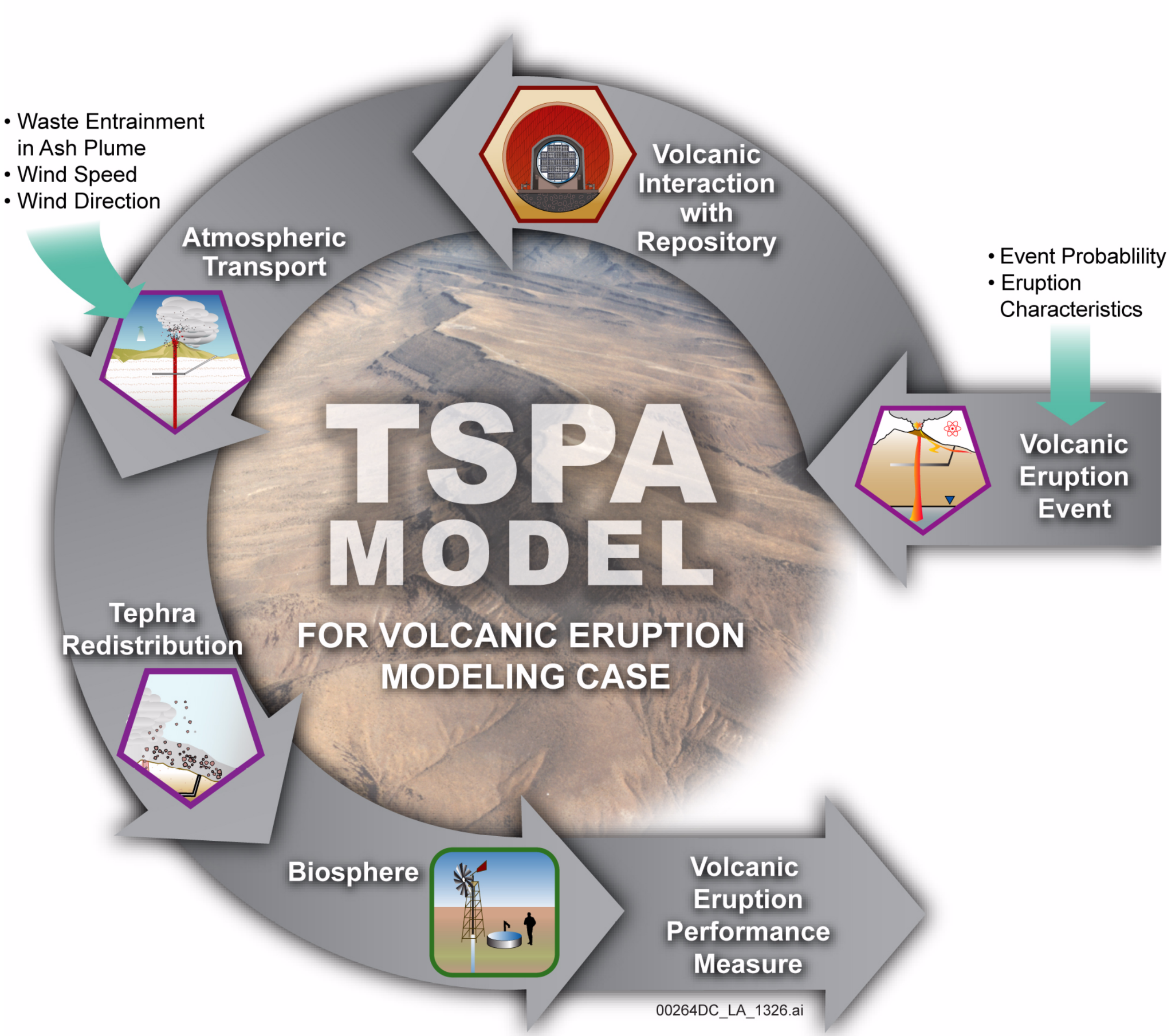


Figure 8. TSPA Model Components for the Volcanic Eruption Modeling case (DOE 2008, SAR Chapter 2; Figure 2.4-6).

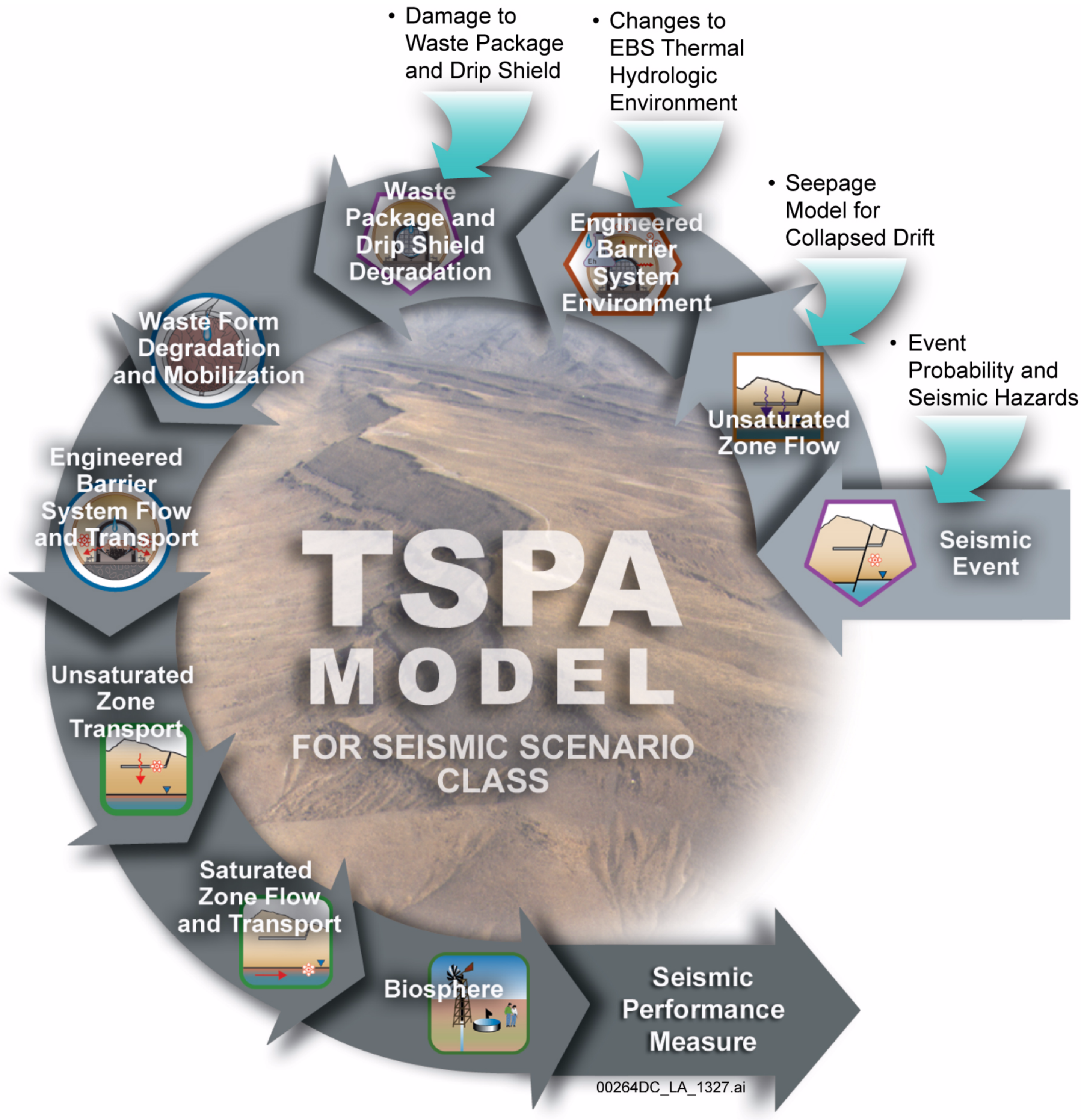


Figure 9. TSPA Model Components for the Seismic Scenario case
(DOE 2008, SAR Chapter 2; Figure 2.4-7).

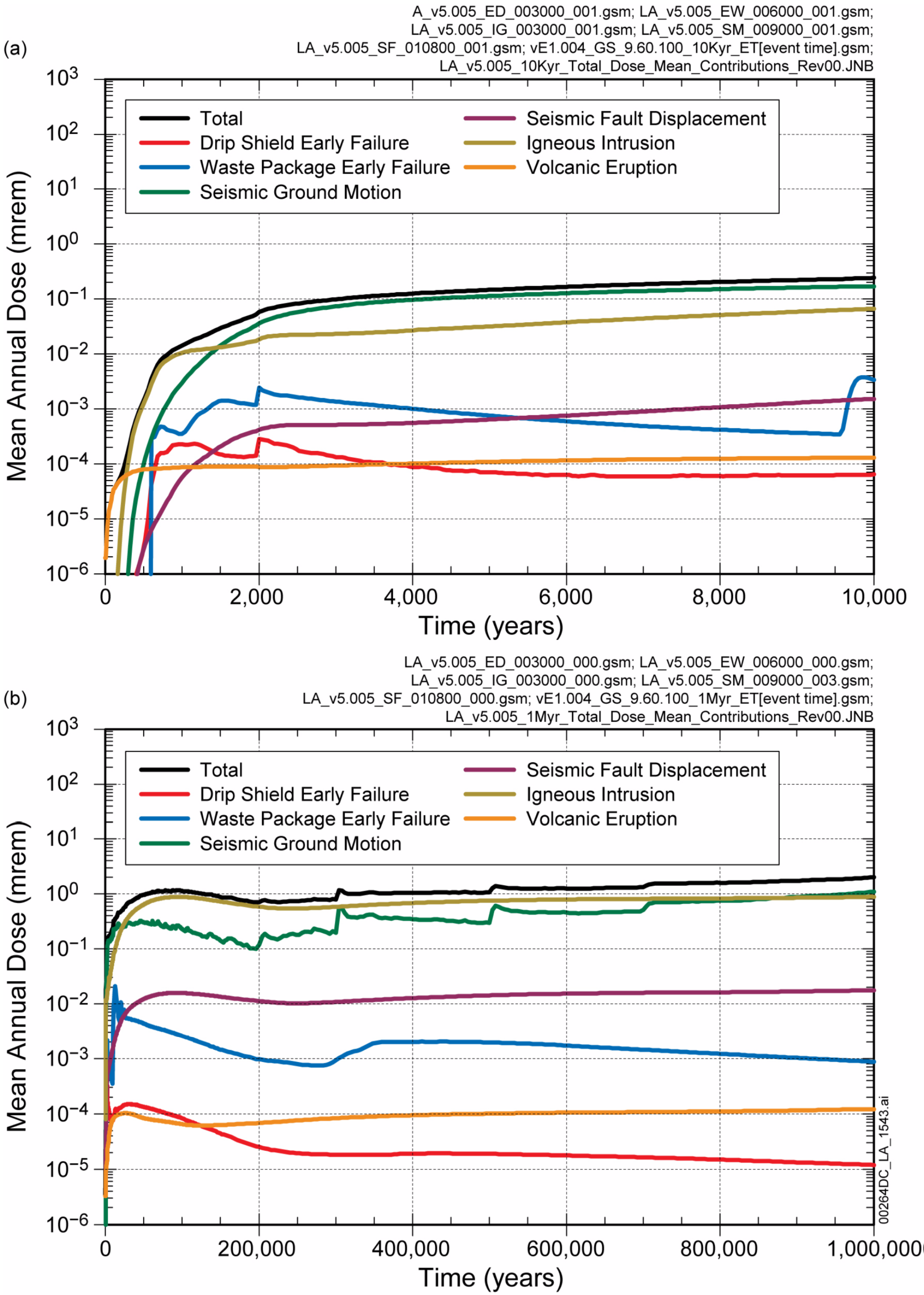


Figure 10. Relative Contributions of Modeling Cases to Total Mean Annual Dose for (a) 10,000 Years and (b) 1 Million Years after Repository Closure (DOE 2008, SAR Chapter 2; Figure 2.4-18).

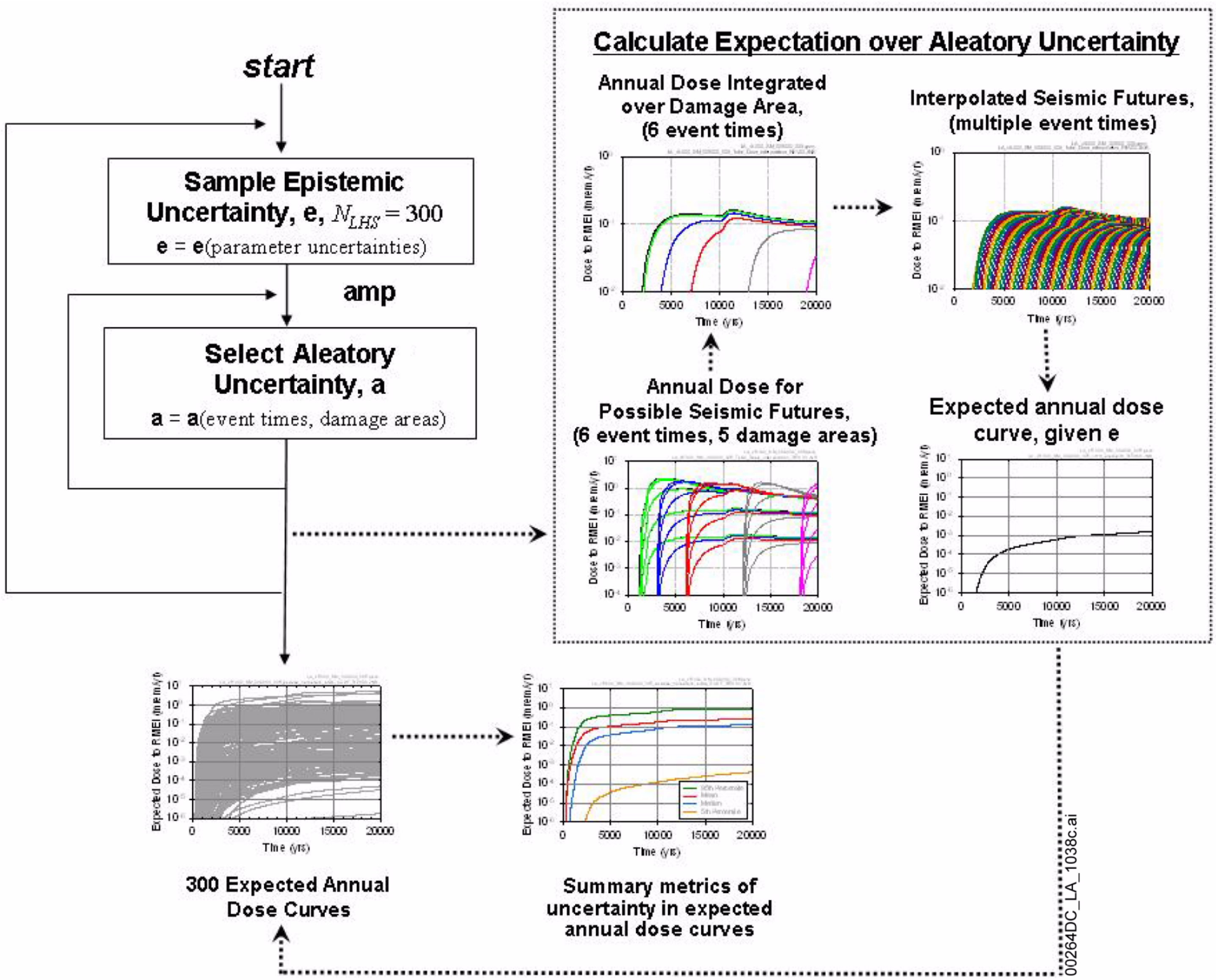


Figure 11. Computational Strategy for Computing the Expected Annual Dose and Associated Summary Metrics for the 10,000-Year Seismic Ground Motion Modeling Case (DOE 2008, SAR Chapter 2; Figure 2.4-8).

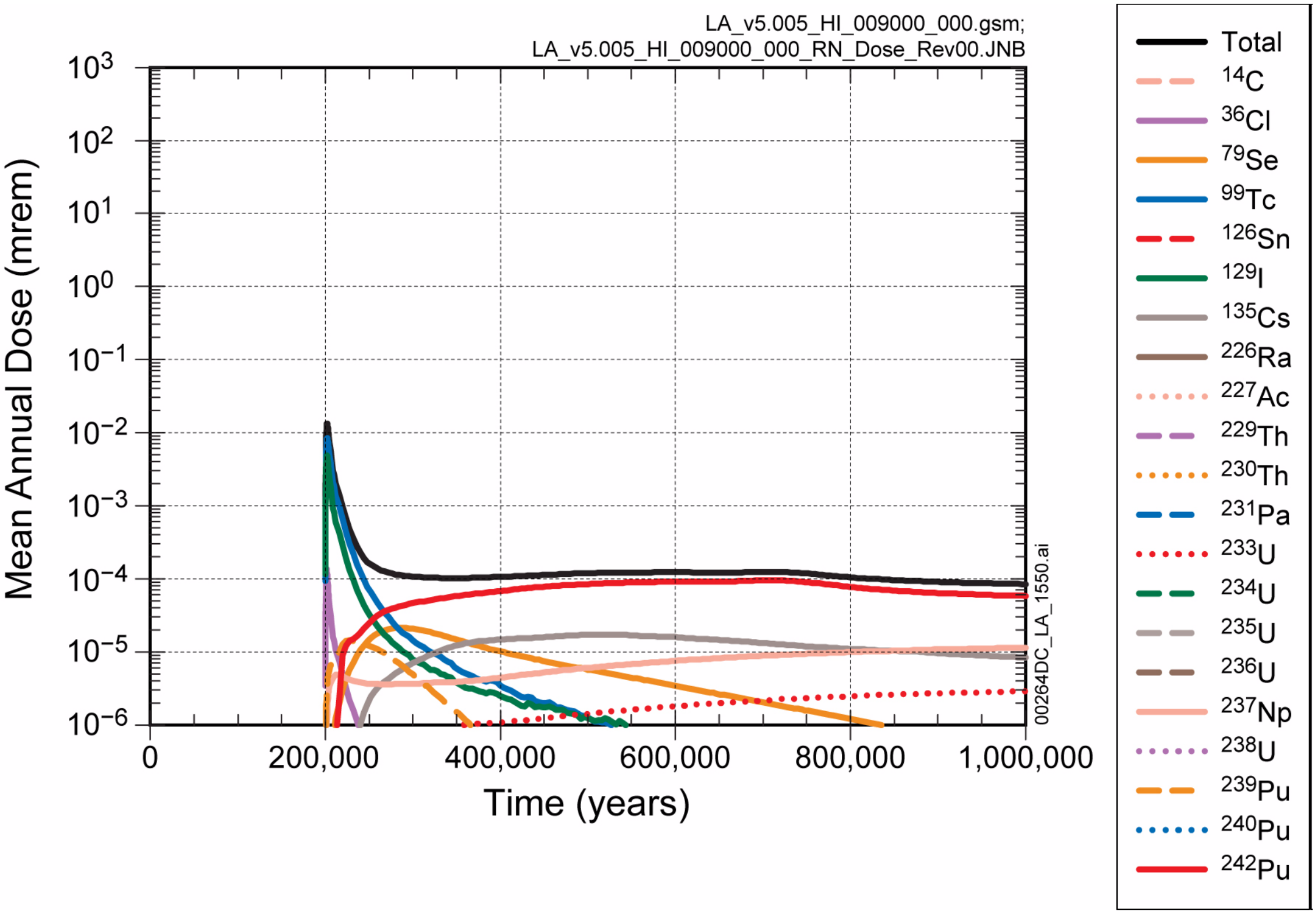


Figure 12. Contribution of Individual Radionuclides to Mean Annual Dose for the Human Intrusion Modeling Case for the Post-10,000 Year Period after Permanent Closure, with Drilling Intrusion Event at 200,000 Years (DOE 2008, SAR Chapter 2; Figure 2.4-159).

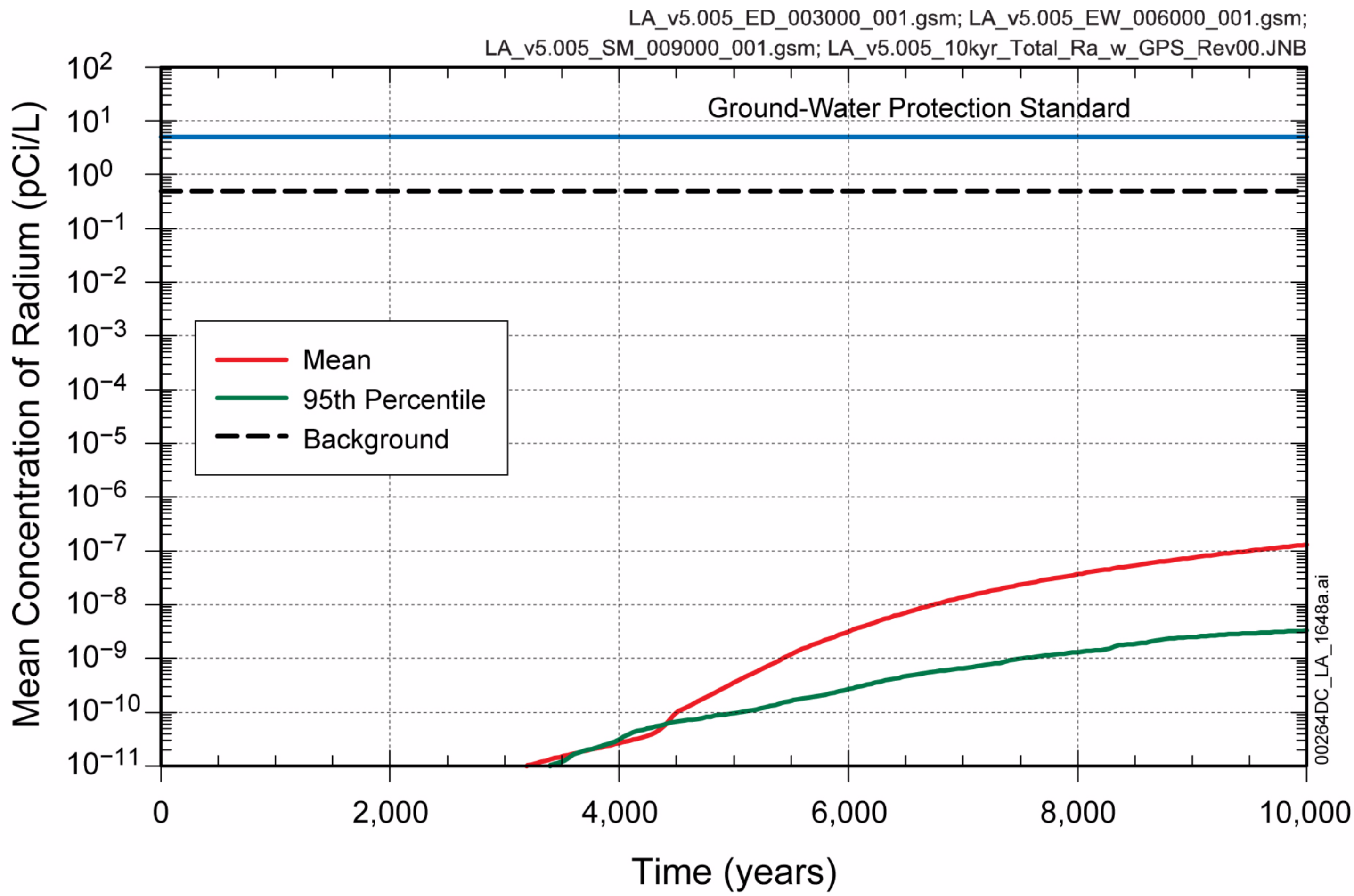


Figure 13. Activity Concentrations for Total Radium (Ra^{226} and Ra^{228}) in Groundwater, Excluding Natural Background, for 10,000 Years after Repository Closure (DOE 2008, SAR Chapter 2; Figure 2.4-12).

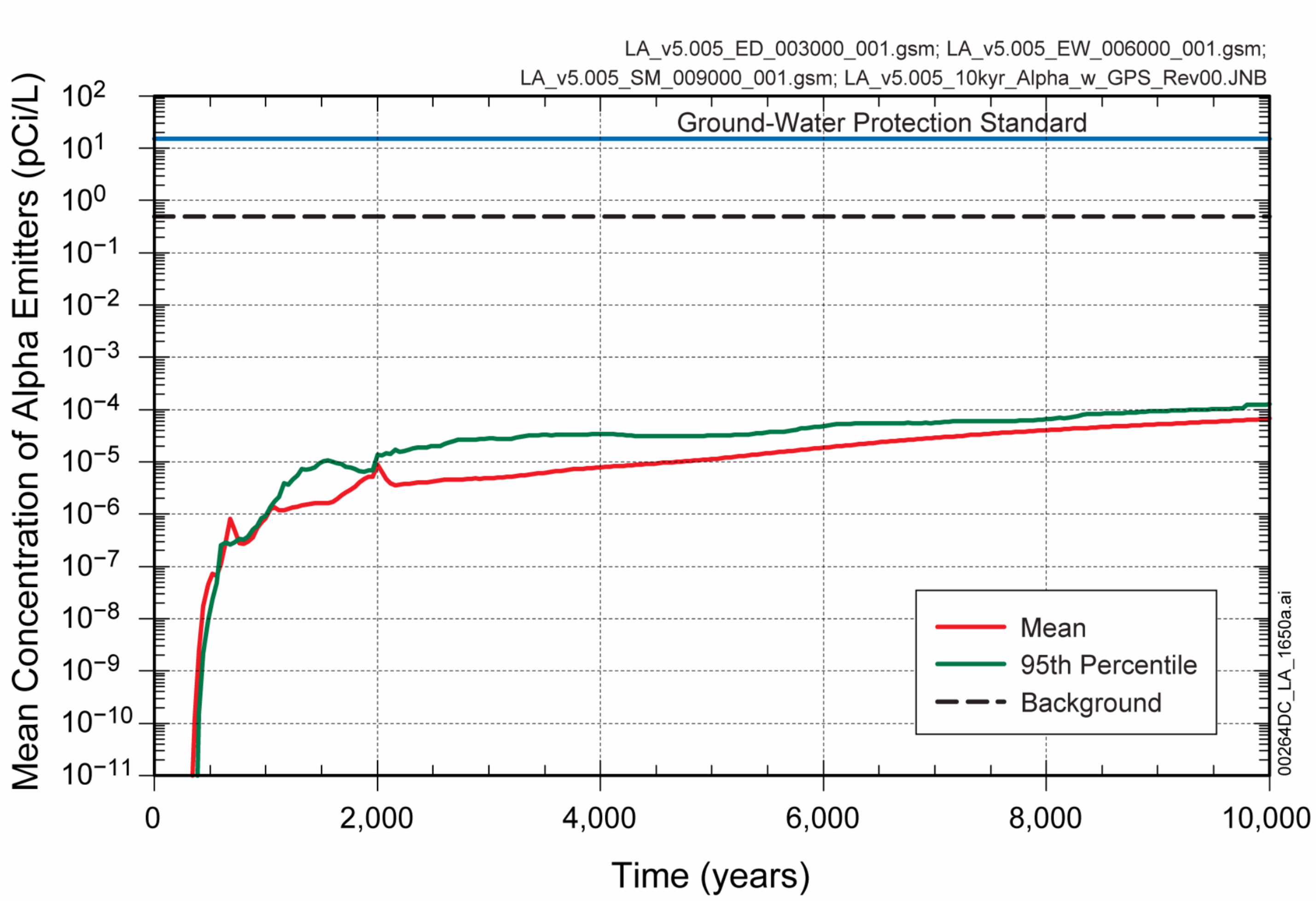


Figure 14. Summary Statistics for Activity Concentration of Gross Alpha (Including 226Ra but Excluding Radon and Uranium) in Groundwater, Excluding Natural Backgorund, for 10,000 Years after Repository Closure (DOE 2008, SAR Chapter 2; Figure 2.4-13).

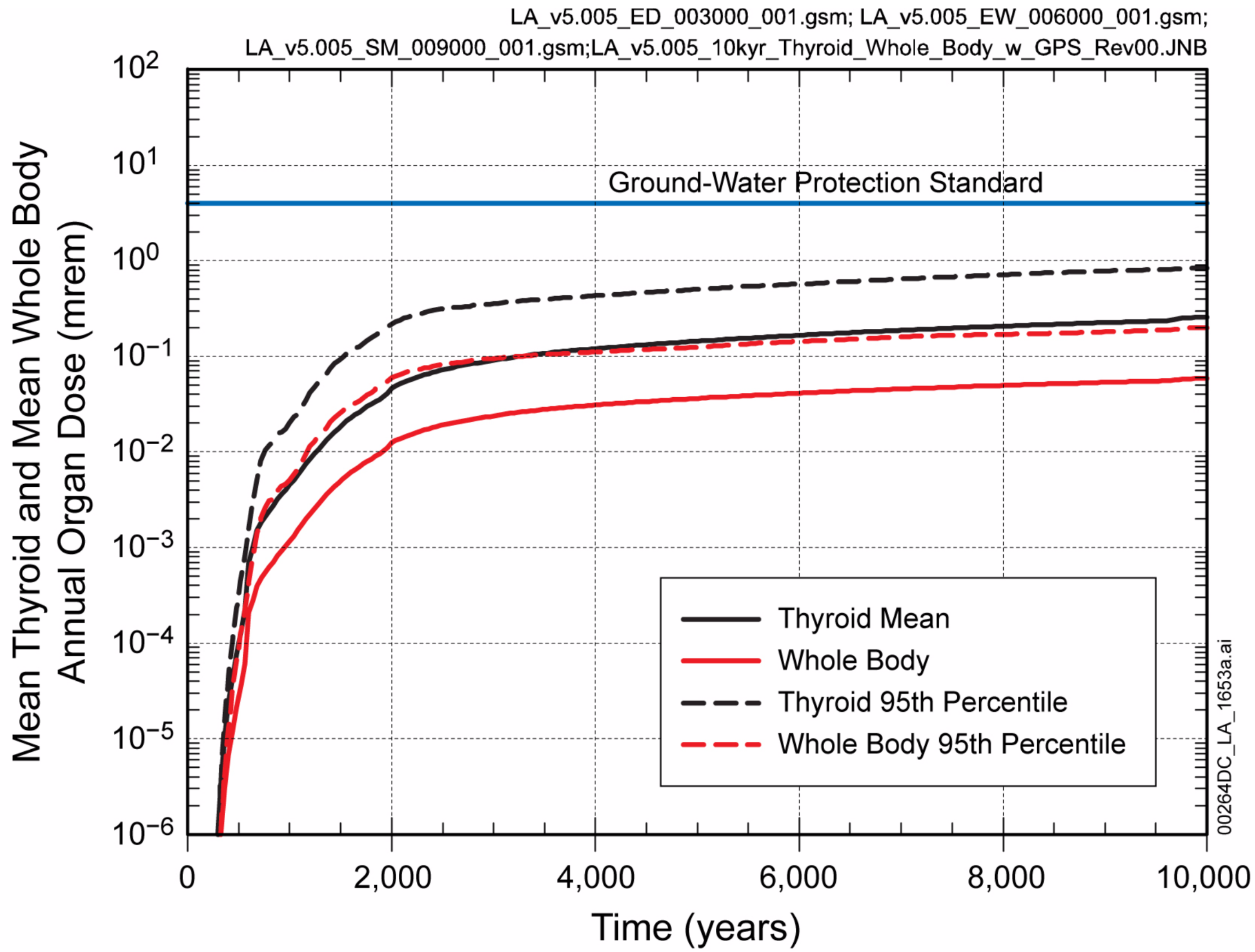


Figure 15. Summary Statistics for Annual Drinking Water Doses for Combined Beta and Photon Emitting Radionuclides, Excluding Natural Background, for 10,000 Years after Repository Closure (DOE 2008, SAR Chapter 2; Figure 2.4-14).

6.0 NRC's Safety Evaluation Report

NRC's Safety Evaluation Report (SER) documenting the staff's review of DOE's license application is detailed in NUREG-1949 Vols 1-5. These volumes include: general information (Volume 1), preclosure safety (Volume 2), postclosure safety (Volume 3), administrative and programmatic requirements (Volume 4), and proposed conditions and license specifications (Volume 5).

This section provides a summary of key aspects of all five volumes of the SER with an emphasis on those aspects that are unique to geologic disposal (e.g., postclosure safety, and underground operations) as compared to those aspects that similar to other NRC regulated activities (e.g., handling of spent fuel canisters at storage facilities, and safety design and procedures for surface facilities).

Following the submission of the license application 12 petitions to intervene in the hearing were filed with Atomic Safety and Licensing Board Panels (ASLBP). Overall, there were 285 admitted contentions (accounting for 13 consolidated contentions) and 21 that were either withdrawn, dismissed, or deemed inadmissible. The admitted contentions comprised 218 contentions related to DOE's Safety Analysis Report and 67 contentions for NEPA related issues (i.e., DOE's Environmental Impact Statement or EIS). The 218 safety contentions predominately addressed postclosure safety, for which there were 173 admitted contentions (12 of the 13 consolidated contentions were in the postclosure phase).

6.1 General Information (NUREG-1949 – SER Volume 1)

> "This is the first volume of the U.S. Nuclear Regulatory Commission (NRC) staff's "Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada." It documents the NRC staff's review and evaluation of general information the U.S. Department of Energy (DOE) provided in its June 3, 2008, license application, as updated on February 19, 2009, that seeks an authorization to begin construction of a repository at Yucca Mountain. In subsequent volumes of the report, Volumes 2-5, the NRC staff plans to present its review and evaluation of the Safety Analysis Report included in DOE's license application.
>
> Consistent with NRC's requirements for the general information, the NRC staff reviewed the following: (i) a general description of the proposed repository, (ii) proposed schedules for repository activities, (iii) a description of security measures, (iv) a description of the Material Control and Accounting Program, and (v) a description of work done to characterize the site." (NRC 2010 Vol. 1, page v)

On the basis of its review and certain DOE commitments (obtained through requests for additional information - RAI), the NRC staff concluded that the regulatory requirements associated with general information [i.e., 10 CFR 63.21(b)(1)-(5)] were satisfied (NRC 2010 Vol.1, page v). The DOE commitments are described in Table 6-1.

Table 6-1. U.S. Department of Energy Construction Authorization Commitments

No.	Description of Commitment	Safety Evaluation Report Reference (Chapter/Section)	RAI Response Reference	Implementation Schedule
1	Update the license application (General Information Figures 1-2 and 1-4 to reflect ownership and the correct acreage of Patent 27-83-0002	1/1.1.3.2.1	DOE 2009	In a future license application update
2	Submission of Physical Protection Plan, compliant with applicable portions of 10 CFR Part 73	3/1.3.3.1; 1.3.3.2.1; and 1.3.3.2.12	General Information Section 3 (DOE 2008)	No later than 180 days after NRC issues a construction authorization
3	Submission of a Material Control and Accounting Program, compliant with applicable portions of 10 CFR Part 74	4/1.4.3.1; 1.4.3.2.1.1; and 1.4.3.2.5	General Information Section 4 (DOE 2008)	No later than 180 days after NRC issues a construction authorization

6.1.1 References

DOE. 2009. "Yucca Mountain-Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 5.8), Safety Evaluation Report Vol. 4, Chapter 2.5.8, Set 1." Letter (May 6) J.R. Williams to F. Jacobs (NRC). Washington, DC: DOE, Office of Technical Management. ML091330698.

DOE. 2008. "Yucca Mountain Repository License Application." Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. ML081560400.

NRC, 2010aa; NUREG-1949, Vol. 1, “Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada; Volume 1: General Information;” August 2010, ML102440298, Washington, D.C.: U.S. Nuclear Regulatory Commission.

NRC’s Safety Evaluation Report (SER)

NRC, 2015a; NUREG-1949, Vol. 5, “Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada; Volume 5: Proposed Conditions on the Construction Authorization and Probable Subjects of License Specifications;” October 2014, ML15022A488, Washington, D.C.: U.S. Nuclear Regulatory Commission.

NRC, 2015; NUREG-1949, Vol. 2, “Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada; Volume 2: Repository Safety Before Permanent Closure;” January 2015, ML15022A146, Washington, D.C.: U.S. Nuclear Regulatory Commission.

NRC, 2014a; NUREG-1949, Vol. 4, “Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada; Volume 4: Administrative and Programmatic Requirements;” December 2014, ML14346A071, Washington, D.C.: U.S. Nuclear Regulatory Commission.

NRC, 2014; NUREG-1949, Vol. 3, “Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada; Volume 3: Repository Safety After Permanent Closure;” October 2014, ML14288A121, Washington, D.C.: U.S. Nuclear Regulatory Commission.

6.2 Preclosure Safety (NUREG-1949 – SER Volume 2)

Volume 2, "Repository Safety Before Permanent Closure," of the SER documents the NRC staff's review and evaluation of the DOE SAR entitled, "Repository Safety Before Permanent Closure," provided by DOE on June 3, 2008, as updated by DOE on February 19, 2009.

> “NRC’s regulations at 10 CFR Part 63 provide site-specific criteria for geologic disposal at Yucca Mountain. Pursuant to 10 CFR Part 63, there are several stages in the licensing process: the site characterization stage, the construction stage, a period of operations, and termination of the license. The multi-staged licensing process affords the Commission the flexibility to make decisions in a logical time sequence that accounts for DOE collecting and analyzing additional information over the construction and operational phases of the repository. The period of operations includes (i) the time during which emplacement would occur, (ii) any subsequent period before permanent closure during which the emplaced wastes are retrievable, and (iii) permanent closure. The license application includes DOE’s subsurface facility development plan (SAR Section 1.3.1) that explains operations in the subsurface facility will be preceded by a period of initial construction, during which three emplacement drifts will be built and commissioned to receive waste. According to DOE, the start of waste emplacement will mark the end of the period of initial construction and the beginning of repository operations in the subsurface facility. DOE stated its plans for the period of operation, also referred to as the preclosure period, is approximately 100 years.
>
> In its review of DOE’s application, the NRC staff used a risk-informed and performance-based review process and considered, among other things, whether the site and design comply with the performance objectives and requirements contained in 10 CFR Part 63, Subparts E and K. In accordance with 10 CFR 63.21, the applicant must include in its SAR a preclosure safety analysis (PCSA). As described in 10 CFR 63.102(f), the PCSA identifies and categorizes event sequences and identifies structures, systems, and components (SSCs) important to safety (ITS) and associated design bases and criteria. The PCSA is part of the risk-informed and performance-based review, which is described further in the following section. An event sequence, as defined in 10 CFR 63.2, is a series of actions and/or occurrences within the natural and engineered components of the facility that could potentially expose individuals to radiation. The applicant’s PCSA must demonstrate that the repository, as proposed to be designed, constructed, and operated, will meet the specified radiological dose limits throughout the preclosure period. The applicant must also demonstrate that the GROA design will not preclude retrievability of the wastes, in whole or in part, from the underground facility where these wastes will be emplaced for permanent disposal (10 CFR 63.111).” (NRC 2015, SER Volume 2, pages xxi and xxii)

The sections of SER Volume 2 were focused on areas that correspond to DOE’s PCSA such as: (1) a site description as it pertains to the PCSA and design of the geologic repository operations area (GROA); (2) description and design information of structures, systems, and components (SSCs); safety controls (SCs); equipment; and operational process activities, both important to safety (ITS) and not important to safety (non-ITS) in the surface and the subsurface facilities of the GROA; (3) identification of hazards and initiating events; (4) identification of event sequences; (5) consequence analysis methodology and demonstration that the repository design meets radiation protection requirements in 10 CFR Parts 20 and 63; (6) identification of ITS SCs and measures to ensure availability and reliability of the safety systems; and (7) proposed design of ITS SSCs and SCs. Additionally, certain sections of the SER addressed aspects of the operational phase in the license application related to (1) the as low as is reasonably achievable (ALARA) program and the Operational Radiation Protection Program (RPP); (2) plans for retrieval and alternate storage, should retrieval become necessary; and (3) plans to facilitate permanent closure and decontamination or the decontamination and dismantlement of the GROA surface facilities.

This summary of the preclosure safety review is organized into discussions of the Surface Facilities and Operations (Section 6.2.1), Subsurface Design and Operations (Section 6.2.2), Retrieval Plans (Section 6.2.3), and Findings and Conclusions (6.2.4). Facilities and operations at the repository surface often would involve the handling of spent fuel and HLW in canisters and as such, these facilities and operations make use of approaches and equipment similar to those employed at other nuclear facilities (e.g., reactors and independent spent fuel storage facilities). Examples include numerous America Society of Mechanical Engineers (ASME) codes and standards and single failure proof cranes for lifting operations. The focus of this Section is on the operations and equipment most relevant to the subsurface rather than the surface facilities, because the surface facilities are not so unique to a repository and commonly occur at other NRC-regulated facilities (e.g., handling of spent fuel canisters at a storage facility).

6.2.1 Surface Facilities and Operations

The main surface facilities include the receipt facility (RF), initial handling facility (IHF), canister receipt and closure facility (CRCF), wet handling facility (WHF), and the aging facility (AF). The NRC review noted consistency of designs with codes and standards (e.g., American Concrete Institute, American Nuclear Society) that are also used for other surface facilities with similar applications, for example shielding due to concrete walls and floors (NRC 2015, SER Volume 2, page 2-40), fire safety systems (NRC 2015, SER Volume 2, page 2-44); ITS diesel generator air start system (NRC 2015, SER Volume 2, page 2-56). Table 7.1 of Volume 2 of the SER provides a listing of the codes, standards, NUREGs, Regulatory Guides, Interim Staff Guidance, and Technical references cited in the review of the DOE’s design of the structures, systems, and components important to safety and their applicability to the geologic repository operations area.

Examples of the NRC review of structures, systems and components (SSCs) important to safety (ITS) that show the relevance of applicable codes, standards and Regulatory Guides are provided below for waste canister transfer equipment, cranes that would be used for lifting and transfer operations, ventilation systems, and fire protection for the surface facilities.

<u>Canister Transfer Machine (CTM)</u>

> “The applicant stated that it used the ASME NOG–1–2004 (ASME, 2005aa) code for the design method of the CTM (SAR Section 1.2.2). The applicant stated that it designed the CTM for DBGM–2 seismic events using the ASME NOG–1–2004 (ASME, 2005aa) codes, and assessed the fragility of the CTM design to show that the CTM has the capacity to perform the safety functions during a seismic event (SAR Section 1.7.1.4).” (SER Volume 3 page 7-38)
>
> “The NRC staff finds that the applicant’s design method for the CTM is acceptable because it is consistent with the ASME NOG–1–2004 (ASME, 2005aa) code for Type I cranes, which is a standard industry practice for the design of cranes in a nuclear power plant, and has design features to minimize the likelihood of a drop of a canister and potential collision of a canister with CTM that may result in radioactivity release. Use of this code at the GROA for the design of cranes is appropriate because the handling of SSCs containing radioactive materials at the GROA are similar to those at a nuclear power plant, as discussed further in Table 7-1. Additionally, the applicant’s seismic fragility assessment of the mechanical handling equipment, such as CTM, is reviewed and found to be acceptable in SER Section 2.1.1.4.3.2.2.” (SER Volume 3 page 7-39)

Cranes

"The NRC staff evaluated the applicant's information on the design method for overhead bridge cranes and jib cranes and finds that the applicant's design methods are adequate because they are consistent with ASME NOG–1–2004 (ASME, 2005aa) for the overhead bridge cranes and ASME NUM–1–2004 (ASME, 2005ac) for the jib cranes. These ASME codes are used in the nuclear industry for design and construction of cranes performing lifting functions. They include overload protection, redundant braking systems, over-travel switches, and protective devices to make the likelihood of a load drop by a crane extremely small. The use of ASME NOG–1–2004 (ASME, 2005aa) Type I cranes is consistent with the design codes and standards for cranes in Section 2.1.1.7.2.3 of the YMRP (NRC, 2003aa). In addition, because the ITS overhead bridge cranes and the jib cranes are SSC ITS (as evaluated by the NRC staff in SER Section 2.1.1.6.3.1) they will be designed to maintain their safety functions under seismic loads, based on DBGM–2 (SAR Section 1.2.2.2.1), as evaluated in SER Section 2.1.1.7.3.1.1.1. The NRC staff's evaluation of site-specific ground motions is documented in SER Section 2.1.1.1.3.5.2.5, where the applicant's analysis was found to be acceptable." (SER Volume 2, page 7-55)

Ventilation

"The applicant stated that its design methods for ITS HVAC systems are in accordance with the codes and standards identified in SAR Section 1.2.2.3. In response to the NRC staff's RAI (DOE, 2009fd), the applicant stated that the HEPA filters measure 610 mm × 610 mm × 292 mm [24 in × 24 in × 11.5 in], which is consistent with the ASME AG–1–2003 (ASME, 2004ac) guidance. In addition, the applicant stated that the design method for the adjustable speed drives (ASDs) is consistent with the National Electrical Manufacturers Association (NEMA) ICS 7–2006 (NEMA, 2006ab). The applicant stated that prefilters and high-efficiency filters for air handling units were designed according to ASHRAE 2004 (ASHRAE, 2004aa), with their efficiency calculated using ANSI/ASHRAE 52.1–1992 (ASHRAE, 1992aa), and that sizing criteria for filters and coils are consistent with ASHRAE 2005 (ASHRAE, 2005aa). In addition, the applicant stated that it sized ducts to maintain a fluid velocity of 12.7 m/s [41.7 ft/s], thereby minimizing particulate settlement consistent with DOE–HDBK–1169–2003 (DOE, 2003ae)." (SER Volume 2, page 7-49)

Fire Protection

"The NRC staff finds that the applicant's design methods and analyses of the ITS fire protection systems are acceptable because they are consistent with (i) Regulatory Guide 1.189 (NRC, 2009ac); (ii) the NFPA 13 (NFPA, 2007ab) standard for the design, installation, inspection, testing, and maintenance of various sprinkler systems (including DIPA systems); and (iii) NFPA 72 (NFPA, 2006aa) standard on the design, inspection, testing, and maintenance standards for various fire alarm and sprinkler monitoring systems. The fire protection program elements (fire hazard analysis, compensatory measures) outlined in RG 1.189 are applicable to all nuclear facilities, including the GROA. NFPA 13 and 72 are nationally-accepted standards used for safety-related fire protection systems in the nuclear industry. The use of these standards is consistent with the design codes and standards for fire protection systems in YMRP (NRC, 2003aa), and the NRC staff finds their use at the GROA acceptable, as proposed by the applicant, and further described in Table 7-1." (SER Volume 2, page 7-95)

6.2.2 Subsurface Design and Operations

Design, construction, and ongoing access and stability are important factors in the safety of preclosure operations associated with emplacement of the waste packages in the repository drifts. Design, maintenance, and operation of the equipment needed to emplace waste packages and drip-shields [e.g., the transport and waste emplacement vehicle (TEV), crane rails to operate the TEV, drip shield emplacement gantry (DSEG), and remote-controlled inspection vehicle] also must be considered.

Drift Stability

Operations in the subsurface relies on the stability of the drift over the 100-year operational period that includes the consideration of drift degradation processes as well as the impact of disruptive events (e.g., faulting and seismicity).

> "The applicant indicated that drift degradation processes included drift degradation, fracturing–fractures (stress-induced fractures), rock deformation, and rockbursts. The applicant assessed the potential degradation of the emplacement drifts during the preclosure period and concluded that drifts will be stable without ground support, based on the drift design (BSC, 2007ai). The applicant further stated that drifts could have spalling of the rock wall; however, such spalling will be mitigated by including a perforated stainless steel liner in the ground support system (DOE, 2009ed). The applicant concluded that these hazards would not cause any adverse effects on the GROA facilities during the preclosure period (BSC, 2008ai)." (SER Volume 2, page 3-13)
>
> "DOE conducted a Probabilistic Fault Displacement Hazard Assessment (PFDHA) within the Probabilistic Seismic Hazard. Fault displacement is a potential hazard to the subsurface GROA because it could damage or shear drifts or waste packages, trigger rockfall within the drifts and shafts, degrade drift walls and ground-support systems, or degrade other components of the engineered barrier system." (SER Volume 2, page 1-27).
>
> "[b]ased on the NRC staff's observation of DOE's elicitation workshops and review of materials produced during the PFDHA, the NRC staff finds that the applicant's process was acceptable. The NRC staff also has extensive knowledge gained through experience evaluating geological evidence for recurrence and slip rates of faults that the applicant considered in the PFDHA (e.g., Stamatakos, et al., 2003aa). Based on this knowledge, the NRC staff concludes that the PFDHA captured the current scientific understanding of probabilistic fault displacement analyses and that the results represent the center, body, and range of viable interpretations, including uncertainty." (SER Volume 2, page 1-28).
>
> "DOE investigated the geological, geophysical, and seismic characteristics of the Yucca Mountain region to obtain sufficient information to estimate how the site would respond to vibratory ground motions from earthquakes. The applicant conducted an expert elicitation on PSHA in the late 1990s (CRWMS M&O, 1998aa) on the basis of the methodology described in the Yucca Mountain Site Characterization Project (DOE, 1997aa). The NRC staff observed all expert elicitation meetings and reviewed summary reports of those meetings as they were produced." (SER Volume 2, page 1-31).
>
> "The NRC staff reviewed the applicant's PSHA input data and interpretations, as described in SAR Sections 1.1.5.2 and 2.2.2.1.1, references therein, and responses to RAIs. The NRC staff concludes that the applicant adequately developed the geological, geophysical, and seismological information necessary to support the expert elicitation. This conclusion is based in part on the NRC staff's evaluations in NUREG–1762 (NRC, 2005aa), where the NRC staff found that the applicant's information was consistent with site conditions at Yucca Mountain. This conclusion is also based on the NRC staff's first-hand knowledge of the geology and seismic characteristics of the Yucca Mountain region, which includes more than a decade of independent geological and geophysical research and study (e.g., Ferrill, et al., 1996aa,ab; Stamatakos, et al., 1998aa; Waiting, et al., 2003aa; Gray, et al., 2005aa; Biswas and Stamatakos, 2007aa). The NRC staff also finds that the resulting suite of ground motion hazard curves; horizontal and vertical components of peak acceleration (defined at 100 Hz); spectral accelerations at frequencies of 0.3, 0.5, 1, 2, 5, 10, and 20 Hz; and PGV are adequate because they are consistent with NRC guidance in Regulatory Guide 3.73 (NRC, 2003ae) and Regulatory Guide 1.208 (NRC, 2007ah). Although these regulatory guides were developed for other types of NRC-regulated facilities (e.g., nuclear power plants and interim spent fuel storage facilities), they are applicable here because (i) the seismic hazard assessment is independent of the type of potentially affected facility and (ii) the methodologies and conclusions in these Regulatory Guides are generally applicable to analogous activities proposed for the GROA.

The NRC staff also reviewed additional geological, geophysical, and seismological information in Wernicke, et al., (2004aa) and Hanks, et al., (2013aa), which were developed after the DOE PSHA elicitation was performed. Wernicke, et al., (2004aa) provided updates to the Global Positioning Satellite (GPS) data for Yucca Mountain to include data from a continuously operating network. These results showed that the anomalously large crustal strain rates indicated by GPS results (Wernicke, et al., 1998aa) considered in the PSHA were in part transient strains associated with the 1992 Little Skull Mountain earthquake and not indicative of increased seismic hazard at the site. Results in Hanks, et al. (2013aa) are based on two studies: one on the physical limits of ground velocities that the lithology at Yucca Mountain could have experienced since deposition, based on the physical limits of rock strength, and a second detailed analysis of the age, distribution, and geometries of precariously balanced rocks along the steep hill slopes in the Yucca Mountain region. Both the physical limits and precarious rock studies in Hanks, et al. (2013aa) suggest upper limits on the amplitudes of earthquake ground motions that occurred in the geological past at Yucca Mountain. These results, thereby, constrain the upper limits of the PSHA at low annual exceedance probabilities and suggest that extremely large ground motions at low annual exceedance probabilities in the DOE PHSA are conservative. These new results, therefore, further support the NRC staff's conclusion that DOE's probabilistic seismic hazard analysis input data and interpretations are adequate. On the basis of its detailed understanding of the Yucca Mountain geology, the NRC staff concludes that new geological and seismological information would not substantially alter the PSHA results, with the exception of over estimation of ground motions at low annual exceedance probabilities, which is described in the following section regarding conditioning of low probability ground motions." (SER Volume 2, page 1-32)

Drift Design and Maintenance

"The NRC staff reviewed the methods the applicant proposed to excavate the underground openings in the accessible areas and selection of the ground support system. The NRC staff also reviewed the construction materials the applicant proposed to use for steel ground support, grout for fully grouted rock bolts, and shotcrete. The NRC staff finds that the applicant's description and design information for the underground openings in the accessible areas are adequate because (i) the excavation methods the applicant selected will minimize construction damage to the surrounding rock and thereby enhance stability of the openings; (ii) the applicant will use well-established empirical methods to select the ground support system (SAR Section 1.3.3.3); and (iii) the applicant will select materials for steel ground support, grout for fully grouted rock bolts, and shotcrete, in conformance with established industry standards (SAR Section 1.3.3.3.3). The NRC staff finds that the applicant's descriptions of the design, monitoring, and maintenance plans are adequate because the applicant (i) described that the accessible openings of the subsurface facility (North Portal, North Ramp, access mains, entrance to the turnouts, intake shafts, and the performance confirmation observation drift) will be designed consistent with the applicant's assumptions in the PCSA regarding the geometry and serviceability of the openings during the preclosure period; (ii) will use design approaches that are used in underground mines and in the tunneling industry; (iii) described how the excavations would be performed (i.e., horizontal openings will be excavated using tunnel-boring machines and vertical openings with raise-boring machines); (iv) explained that it will monitor the performance of the accessible openings through regular visual inspection by qualified personnel and will implement a geotechnical instrumentation program to measure drift convergence, ground support loads, and potential overstressed zones; and (v) stated the monitoring and maintenance program will be performed using methods similar to those used in underground openings in civil and mining industries." (SER Volume, page 2-128)

"The effectiveness of rock bolts to anchor surface-protection ground support elements could be undermined if the rock bolts corrode during the preclosure period. The applicant expects stainless steel rock bolts to perform better than carbon steel rock bolts because the stainless steel material will be less susceptible to general corrosion than carbon steel. In response to an NRC staff RAI, the applicant also stated that the environment in the drift will not be conducive to stress corrosion cracking (SCC) of stainless steel rock bolts and that confirmatory studies and tests are planned to verify that SCC will not occur (DOE 2009gu). The NRC staff notes that for SCC to occur, the relative humidity needs to be high enough to have sufficient aqueous environment but also be low enough to have sufficient chloride concentration and no drying out of salts. Given the thermal load and ventilation planned for the drift through the preclosure period, the NRC staff finds that the applicant's statement that the environment in the drift will not be conducive to SCC is acceptable. The relatively higher humidity inside the rock bolt boreholes and the localized stress concentrations in the void zone in the lithophysal rock unit may result in localized SCC of the rock bolt. However, the NRC staff finds that this would have a limited impact, as the openings are expected to be stable without ground support." (SER Volume 2, page 2-131)

"To maintain the functional requirements of emplacement drifts, including emplacing drip shields at the end of the preclosure period, the applicant proposed a monitoring and maintenance plan (DOE 2009ea,ef,gk). The applicant's descriptions of monitoring and maintenance plans are acceptable because DOE described how it will (i) monitor rock wall convergence at preselected locations along the openings using convergence pins attached to the rock or fixed laser targets attached to the head of rock bolts, (ii) monitor the deformation of the stainless steel liner using laser scanning at additional selected locations, and (iii) use the convergence data and other available information to determine the need for maintenance to preserve the equipment operating envelopes and meet operational needs. The ability to perform monitoring in these inaccessible areas will be contingent on availability of power and communications provisions enabling remote inspection and observation. Power, communications, and vehicle SSCs required for remote monitoring and maintenance in inaccessible areas are evaluated in SER Sections 2.1.1.2.3.2.3 and 2.1.1.2.3.6.2." (SER Volume2, page 2-131)

Subsurface Thermal Management and Ventilation

"The applicant will design a forced air subsurface ventilation system to remove heat from the emplaced waste and maintain temperature limits in the drift, as listed in SAR Tables 1.3.1-2 and 1.3.5-2, and to provide fresh air to personnel and equipment. The subsurface ventilation system components will include fans, isolation barriers, airflow regulators, access doors, and instrumentation for controlling and monitoring the system. An interconnected system of subsurface openings that will consist of intake ramps, access and exhaust mains, access turnouts, emplacement drifts, intake and exhaust shafts, and shaft access drifts will be utilized to circulate ventilation air. The ventilation system location and functional arrangement were described in SAR Section 1.3.5.1.2. The function of specific system components and their design was described in SAR Section 1.3.5.1.3. In SAR Section 1.3.5.1.3.2, the applicant described the operation of the ventilation system during simultaneous emplacement and development in which isolation barriers will be used to direct airflow in the desired direction. SAR Figure 1.3.5-5 showed the ventilation system layout after full emplacement, and SAR Figures 1.3.5-6 and 1.3.5-7 highlighted ventilation system operation during concurrent emplacement and development. The description of the airlock system and isolation barriers that will isolate (i) inlet airflow from exhaust airflow and (ii) the emplacement area from the development area was provided in SAR Section 1.3.5.1.3.2. The applicant plans to provide a nominal airflow rate of 15 m3/s [32,000 cfm] in each emplacement drift with thermal loading of up to 2.0 kW/m [0.61 kW/ft] and, if required, will be able to vary the drift airflow rate between 0 and 47 m^3/s [0 and 100,000 cfm]." (SER Volume 2, page 2-125)

"The applicant described the subsurface ventilation system maintenance considerations in SAR Section 1.3.5.1.5. It asserted that ventilation fans will be monitored and maintained according to manufacturer guidelines, and the fans will be located on the surface, providing easy access for maintenance. According to the applicant, emplacement access doors will require regular periodic inspection with the bulkhead and frame requiring minimal maintenance. The applicant stated that emplacement door components will have a modular design that facilitates easy replacement. The applicant does not plan any routine maintenance activities for door actuators, which will be remotely monitored and replaced, if necessary. The applicant also anticipates that emplacement door actuators will operate only a few hundred times, as approximately 100 waste packages will be emplaced per drift. SAR Section 1.3.5.3.2 presented an analysis of thermal effects under off-normal conditions, such as ventilation shutdown. The applicant considered three different cases: (i) analysis of complete ventilation shutdown in the absence of natural convection, (ii) naval SNF behavior under ventilation shutdown with natural convection, and (iii) thermal effect of drift obstruction. In the first analysis, the applicant demonstrated that waste package components will not reach their temperature limit within 30 days after loss of ventilation, as shown in SAR Figures 1.3.5-17 and 1.3.5-18. The thermal analysis of naval SNF, considering only natural convection, showed that the waste package temperature will be below values mentioned in SAR Table 1.3.1-2. The applicant also stated that the probability of an emplacement sequence within the drift, where a naval SNF waste package [(12.9 kW) (12.2 Btu/sec)] will be placed beside one with commercial spent nuclear fuel (CSNF) having the limiting thermal load 18.0 kW (17 Btu/sec), is extremely small. In the third analysis, the applicant showed that the ventilation system will be capable of maintaining normal airflow with 94 percent localized blockage of a single emplacement drift. The applicant also stated that any potential rockfall during the preclosure period in the lithophysal and nonlithophysal rock will be prevented by the perforated stainless steel sheet and rock bolts of the ground support system." (SER Volume 2, page 2-126)

"The NRC staff reviewed the subsurface ventilation system maintenance considerations using the guidance in YMRP Sections 2.1.1.2 and 2.1.1.7.3.3(II). The NRC staff reviewed the descriptions of the maintenance activities for the ventilation fans and emplacement access doors. The NRC staff also reviewed the design features that will facilitate maintaining the subsurface facility ventilation system. Additionally, the NRC staff reviewed the applicant's analysis of thermal effects, in the event of ventilation shutdown. The NRC staff finds that the applicant's description of the subsurface ventilation system maintenance considerations are adequate because the applicant (i) explained that ventilation fans will be monitored and maintained according to manufacturer guidelines, and the fans will be located on the surface, providing easy access for maintenance; (ii) explained that emplacement access doors will be regularly inspected, while the bulkhead and frame will receive minimal maintenance; (iii) explained that emplacement door components will have a modular design that facilitates easy replacement and anticipates that emplacement door actuators will operate only a few hundred times, as approximately 100 waste packages will be emplaced per drift; and (iv) used standard techniques and tools to perform the thermal analyses that determined that the ventilation system will be capable of maintaining normal airflow with 94 percent localized blockage of a single emplacement drift." (SER Volume 2, page 2-127)

Transport and Waste Emplacement Vehicle (TEV)

"The applicant described the TEV as a rail-based, self-propelled, multiwheeled vehicle designed for transporting waste packages from the surface facilities (CRCFs and IHF) to the subsurface emplacement areas of the repository. The applicant categorized five main TEV functions: (i) handling the waste packages on associated pallets in the surface facilities by performing docking, lifting, and lowering maneuvers; (ii) providing waste package radiation shielding to personnel in unrestricted areas; (iii) transporting waste packages from the surface facilities to the subsurface facility in a controlled and safe manner; (iv) lifting, lowering, and positioning the waste package during the emplacement process in the drift; and (v) safely returning the TEV to the surface facility. The applicant also proposed to use the TEV for retrieval operations, if needed, by reversing the emplacement operations. The applicant emphasized that even though the TEV is a one-of-a-kind transportation system, its construction, material, and functions are considered similar to those of mining equipment and gantry cranes in the nuclear industry." (SER Volume 2, page 2-74).

The NRC review considered DOE's analysis of a potential TEV accident (i.e., impact collision between the TEV and an emplaced waste package and thermal analysis predicting the temperature within the TEV during emplacement operations (note the TEV is operated remotely).

> "The applicant indicated that the primary human interactions during subsurface operations will involve communication and control of the TEV by operators in the control center. The applicant described the use of high-intensity lights and a camera onboard the TEV to provide feedback to operators in the control center. The applicant also described the use of programmable logic controllers (PLCs) that will accept initiating commands from operators to execute predefined, preprogrammed instructions and maneuvers." (SER Volume 2, page 2-108).
>
> "For the initiating event 'TEV Impact During Transit,' the applicant indicated that collision with another object could take place if the TEV is a runaway while traversing the North Ramp, leading to a derailment and impact with the tunnel wall, or if the TEV collides with an object along the rail line, as outlined in the ESD SSO–ESD–02 in BSC (2008bj, Table 11) and BSC (2008bk, Section B1.4.4). Using this facility-specific information, the applicant developed the fault tree model for the initiating event. The applicant identified three potential failure modes: (i) another vehicle being driven into the TEV on the surface, (ii) uncontrolled descent of the TEV down the North Ramp resulting in an impact with the tunnel wall, and (iii) TEV impact with another object along the rail line due to either spurious signal from the drive controllers or failure of the manual control switch (BSC, 2008bk). The fault tree comprised three subfault trees, one for each failure mode." (SER Volume 2, page 3-64)
>
> "The NRC staff reviewed the information the applicant used to determine the annual frequency of occurrence of the initiating event 'TEV Impact During Transit,' provided in SAR Table 1.6-3 and BSC (2008bk). The associated fault tree model is provided in BSC (2008bk). The applicant used the alpha-factor method (Mosleh, et al., 1998aa) to model the common-cause failure of the speed sensors of the motors. The NRC staff finds that the applicant's conclusion that the frequency of TEV impact during transit will be primarily controlled by an operator driving another vehicle into it, which is approximately 99 percent of the contribution, is acceptable because (i) the human-induced hazard was appropriately identified and (ii) mechanical or electrical failure has a low probability of occurrence, as shown in the fault tree for the initiating event 'TEV Impact During Transit' (BSC, 2008bk). Furthermore, the NRC staff finds that the applicant's proposed actions to reduce the probability of TEV impact during transit are appropriate because these actions include installation of special crossing barricades and signals at all surface intersections, restriction of all traffic within the same area of a loaded TEV in the subsurface facilities, the TEV travel speed limit {roughly 3.2 km/hr [2 mph]}, and activity monitoring by an operator via camera, as described in BSC (2008bk, Table E6.2-2). Therefore, the NRC staff finds that the applicant's estimated annual frequencies for the initiating events are acceptable because they were calculated using appropriate methods and reliability information." (SER Volume 2, page 3-65)
>
> "The NRC staff evaluated the applicant's information on the design criteria and design bases of the TEV and finds that the design bases and design criteria for the TEV are adequate because they (i) include a design criterion to keep the TEV's center of gravity low and providing a wide base to protect against a tipover during a DBGM–2 seismic event; (ii) include the single failure criterion in ASME NOG–1–2004 (ASME, 2005aa) code for Type I cranes that would prevent a load drop as a result of single component failure; (iii) encompass a design criterion to equip the TEV with special drive mechanisms and braking systems to protect against TEV runaway, and the proposed run-away prevention features are consistent with the industry practice of protecting rail-based vehicles from unintended motions; (iv) adopt the use of ASME NOG–1– 2004 (ASME, 2005aa) Type I single-failure proof as a design criterion for protection against the derailment of the TEV during a DBGM–2 seismic event, consistent with the standard engineering practice in nuclear industries for heavy load dropping protection during an accident

(e.g., derailment); (v) address protection of personnel from radiation exposures through the use of interlocks and shielded enclosures, which is consistent with the nuclear industry radiation protections measures; and (vi) include a design criterion to protect the waste packages from ejection through the use of locks to the TEV front and rear shield doors." (SER Volume 2, page 7-63)

Drip Shield

Additionally, the installation of the drip shield represents a unique feature of the proposed repository. DOE explained how the design made use of current technology and it would be installed via the drip shield emplacement gantry (DSEG). In its evaluation, the NRC staff stated the following:

"According to the applicant, drip shields will form a continuous barrier throughout the entire length of the emplacement drift by interlocking the drip shield segments. The applicant stated that the drip shield will accommodate an interlocking feature to prevent the separation between contiguous drip shield segments and a minimum lift height of 1,016 mm [40 in] will be required to interlock the drip shield segments. Furthermore, the drip shield interlocking feature will include water diversion rings and connector plates that will divert the liquid moisture at the seams between the drip shield segments. SAR Figure 1.3.4-15 detailed the drip shield interlock feature, and in response to an NRC staff RAI, the applicant provided a sequence of isometric sketches that illustrated the drip shield interlocking process and figures that demonstrated the height clearance required to interlock two drip shields (DOE, 2009dr).

The applicant stated that, except for the attachment to the Alloy 22 base, drip shield components will be connected to each other by welding. According to the applicant, the Alloy 22 base plates will be mechanically attached to the titanium components by Alloy 22 pins because titanium and Alloy 22 cannot be reliably welded together. The applicant included codes and standards governing physical and mechanical properties (e.g., density, elongation, yield, and ultimate tensile stresses) of Titanium Grades 7 and 29 in SAR Table 1.3.2-5. In response to the NRC staff RAI (DOE, 2009dr) regarding the codes and standards for the drip shield design and fabrication, the applicant extracted codes and standards as applicable for materials, welding, postweld heat treatment, and postweld nondestructive examination of the drip shield from Yucca Mountain Project Engineering Specification Prototype Drip Shield (BSC, 2007bu). The applicant stated that the codes and standards cited in the prototype specification were adopted from the ASME Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 2001aa) and American Welding Society standards for welding. In addition, the applicant stated that the prototype program will be used to demonstrate and confirm the design suitability and progressively develop and refine the production fabrication process." (SER Volume 2, page 2-100)

"The applicant described the DSEG design as a self-propelled, rail-based crane, which will be similar to the TEV based on nuclear and industrial crane technology. The main components of the DSEG will include (i) a steel frame structure capable of supporting the weight of a drip shield; (ii) a lifting system composed of four lifting brackets, screw jacks, shot bolts, and gantry motors that can vertically lift the drip shield for transportation; (iii) a self-propulsion system containing electric drive motors with integrated disk brakes and fail-safe capabilities; (iv) an onboard programmable logic controllers (PLCs) network that communicates with the CCCF and with thermal and radiological sensing instrumentation onboard the DSEG; (v) an electrified third rail supplied by a dual-power-pickup mechanism to provide power to onboard electrical systems; (vi) air-conditioned cooled electronic cabinets to protect temperature-sensitive equipment; (vii) a fire-suppression system that detects and automatically operates when needed; and (viii) instrumentation and control system (I&C) containing articulated cameras, ultrasonic sensors, and high-intensity lights. The applicant provided a more extensive discussion on the drip shield emplacement operations and its conceptual design, including drive system, electrical and control systems, braking controls, cooling system, vision system, thermal and radiation monitoring system, fire protection, and communication systems in a supplemental document (BSC, 2007cf)." (SER Volume 2, pages 2-79 and 2-80)

"The applicant considered the operations needed to install drip shields over the waste packages in the emplacement drifts, as shown in BSC (2008bj, Figure 16). The applicant described its plans to install the drip shields toward the end of subsurface operations and prior to permanent closure of the repository. The applicant described the subsurface SSCs for the underground openings and the invert structures and rails, power distribution infrastructure, and subsurface ventilation, which functions within the serviceability limits needed for subsurface operations through the preclosure period [BSC (2008bj, Attachments A and B)]. SAR Sections 1.3.3.3.2 and 1.3.4.4.2 stated that the applicant will use monitoring and inspection programs to assess the need for and frequency of maintenance of the subsurface structures and systems. In an RAI, the NRC staff requested the applicant to clarify its approach to preventing or mitigating potential event sequences related to subsurface structures or systems failure, such as (i) failure of the invert structure due to corrosion, thermal expansion, or loss of rock support; (ii) collapse of an emplacement drift, exhaust main, or exhaust shaft; (iii) loss of operating envelope due to wall convergence; (iv) ventilation failure due to blockage of an exhaust conduit, such as ventilation raise or exhaust main or shaft; or (v) rock deformation due to fault displacement or thermal expansion resulting in buckling or misalignment of the third rail used for power supply or a slotted microwave guide system for communications. In its response to the RAI (DOE, 2009ed), the applicant stated that it established design criteria and bases to ensure stability of the subsurface structures and systems, and a monitoring, inspection, and maintenance program will address any deterioration of the structures and systems in a timely manner. The NRC staff's review of the stability of subsurface structures and systems is documented in SER Section 2.1.1.2.3.7."
(SER Volume 2, pages 4-15 and 4-16).

6.2.3 Retrieval Plans

"The applicant proposed a preclosure period of 100 years, which includes construction of the geologic repository operations area, emplacement of waste in the underground facility, performance confirmation, and the 50-year retrievability period prescribed in 10 CFR 63.111(e) (SAR Section 2.2 and general information Section 1.1.2.1). The applicant's retrieval plan consists of maintaining access to waste packages in emplacement drifts throughout the preclosure period, such that waste packages could be retrieved, if necessary, by reversing the operational procedure used for waste emplacement. The applicant plans to accomplish this by (i) designing the ground support system in the access and ventilation mains and emplacement drifts to function for 100 years; (ii) developing a maintenance plan to test, inspect, and repair ground support as necessary to ensure functionality of the underground openings through a 100-year preclosure period; and (iii) designing the subsurface communication and transportation infrastructure to function through the preclosure period to support access for maintenance or equipment replacement as needed. The applicant also stated that if off-normal events (i.e., those outside the bounds of routine operations but within the range of analyzed conditions for SSCs), such as collapse of an emplacement drift section occurred, specialized procedures and equipment could be developed to restore access to waste packages. The applicant also identified an alternate storage facility location. The applicant did not propose the option of backfilling of emplacement drifts." (SER Volume 2, page 9-2)

"The applicant described a monitoring and maintenance plan for the ground support system to keep the subsurface facility openings stable to permit access to the SSCs and waste packages. The applicant's monitoring plan for accessible openings (such as access mains and the North Ramp) consists of regular visual inspection of the openings by qualified personnel and use of a geotechnical instrumentation program to obtain measurements of drift convergence, ground support loads, and potential overstressed zones (DOE, 2009bb). The applicant indicated that, for the emplacement drifts and turnouts, it will use remotely operated equipment to inspect the openings to detect any indications of rockfall, drift deterioration, or instability and to measure drift convergence at locations selected on the basis of previous inspections (DOE, 2009bb). The applicant stated that every emplacement drift and turnout will be inspected over its entire length: once a year initially after waste emplacement but at a modified frequency subsequently. The applicant stated that subsequent inspection frequencies would use results of previous inspections and geologic mapping to support any changes because the frequency of monitoring is a key component of the monitoring program." (SER Volume 2, page 9-3)

"The NRC staff compared the emplacement operations to the retrieval operations and determined that these operations are the same except that during retrieval, the transport and emplacement vehicle (TEV) must climb a 2.5 percent grade when loaded with a waste package. During emplacement, the TEV is only loaded when descending. The NRC staff reviewed the TEV design to determine whether the TEV is designed to perform retrieval operations and whether the loading system (or propulsion duty cycle) is designed to climb a 2.5 percent grade when loaded with a waste package. The NRC staff's review of the TEV design in SER Section 2.1.1.7.3.5.1 finds that the TEV is designed to support waste package transportation and its drive system is designed to negotiate a 2.5 percent grade in both downward and upward directions when loaded with a waste package. As discussed in SER Section 2.1.1.2, the invert structure and rails, electrical power system, communication system, and drift ventilation system are designed to support retrieval operations. The NRC staff finds that the underground facility design along with the monitoring and maintenance programs would ensure accessibility to waste packages throughout the preclosure period.

The NRC staff also evaluated whether the applicant's plan to inspect the emplacement drifts and turnouts using remotely operated equipment is reasonable. In response to NRC staff's RAI, the applicant stated that it will inspect the entire length of every emplacement drift and turnout annually. After reviewing the applicant's RAI response (DOE, 2009bb), the NRC staff concludes that the applicant has provided sufficient spatial and temporal coverage of observations necessary to assess performance of the ground support systems. The applicant stated in DOE (2009bb) that it might change its inspection frequency if information gathered up to that point in time supports such a change. The applicant stated that the basis for change in the inspection frequency of ground support would be properly documented and supported as required by 10 CFR 63.44(c). The staff concludes that the applicant could adjust the temporal frequency of inspections as conditions change in accordance with the 10 CFR 64.44 process, provided the inspection is frequent enough to permit an assessment of the rate of any change in ground support conditions. The NRC staff's review of DOE's commitment to use the 10 CFR 63.44 process is documented in SER Section 2.5.10.1.3.1.1. As the applicant would document the basis for changes in the inspection frequency of ground support, the NRC staff finds that it can evaluate the effects of changes in the frequency of inspection, as needed. The NRC staff finds that the applicant's plan to inspect the emplacement drifts and turnouts once a year initially and modify the inspection frequency as necessary in accordance with 10 CFR 63.44 provides adequate temporal coverage of observations necessary to assess performance of the ground support systems." (SER Volume 2, pages 9-3 and 9-4)

"For an off-normal condition involving rockfall, the NRC staff reviewed the applicant's evaluation of a scenario involving waste package burial in BSC (2008bt) and its response to staff's RAI in DOE (2009ba, Enclosure 4). The NRC staff reviewed the applicant's assumptions for representing a rockfall condition and the analytical approach to determining the thermal effects of rock rubble. The NRC staff finds the applicant's assumptions regarding the rockfall covering the waste package to be reasonable because they approximate the physical conditions after the postulated rockfall. The NRC staff finds the analytical approach the applicant used to be appropriate because the numerical code used (ANSYS) is a widely accepted code for performing such thermal analyses. The NRC staff reviewed the applicant's calculations taking into consideration any reliance on subsurface ventilation for cooling the buried waste packages and potential impacts on the retrieval schedule under off-normal conditions. The NRC staff reviewed the temperature profile of a waste package partially or completely buried in rubble and finds, on the basis of verifications against the information provided in BSC (2008bt), that the applicant's analyses considered appropriate failure modes of concern and the associated temperature limits as per specifications."
(SER Volume 2, page 9-6)

"On the basis of its review of the applicant's description of plans for retrieval and plans for monitoring drift convergence documented in SER Section 2.1.1.2, the NRC staff finds that (i) the applicant's description of the plan for maintaining access to waste packages in emplacement drifts through the preclosure period under normal operating conditions is acceptable because the applicant has provided sufficient information about the feasibility of retrieval plans under normal conditions; (ii) the applicant adequately identified retrieval scenarios under degraded drift conditions because the two off-normal scenarios the applicant analyzed bound the possible range of adverse conditions; (iii) the applicant's retrieval plan description under off-normal conditions is acceptable because it considered potential scenarios that could lead to rockfall and derailment, and the applicant performed analyses using accepted engineering models and codes; and (iv) the applicant's proposed solutions to address off-normal conditions are reasonable because they would be feasible and could be implemented within the proposed repository design concepts." (SER Volume 2, page 9-7)

6.2.4 Conclusions

Volume 2 of the SER concludes:

> "The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed and evaluated the U.S. Department of Energy's ("DOE" or the "applicant") Safety Analysis Report (SAR), Chapter 1: Repository Safety Before Permanent Closure and the other information submitted in support of its license application and has found that DOE submitted applicable information required by 10 CFR 63.21. The NRC staff has also found, with reasonable assurance, that subject to proposed conditions of construction authorization, DOE's design of the proposed geologic repository operations area (GROA) and preclosure safety analysis complies with the preclosure performance objectives at 10 CFR 63.111 and the requirements for preclosure safety analysis of the GROA at 10 CFR 63.112." (SER Volume 2, page 11-1)

The NRC identified 6 proposed conditions of the construction authorization. These were very specific actions and information that would need to be supplied to assure regulatory compliance during the operational period:

(1) Within 90 days of issuance of construction authorization, DOE must confirm its site characterization information and related analyses in the SAR submitted in accordance with 63.21(c)(1) continue to be accurate with respect to (i) site boundaries, (ii) man-made features, (iii) previous land use, (iv) existing structures and facilities, and (v) potential exposure to residual radioactivity. DOE must provide to the NRC written notification when its confirmatory analysis is complete. This notification must include, for NRC staff's verification, a copy of DOE's confirmatory analysis. (SER Sections 2.1.1.1.3.1 and 2.1.1.1.3.9)

(2) DOE shall not, without prior NRC review and approval, accept DOE spent nuclear fuel (SNF) in multicanister overpacks (MCOs) or commercial mixed oxide (MOX) fuel.

Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of these MCOs and MOX fuel at the GROA or (ii) demonstrates, through the PCSA, that MCOs and MOX fuel can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112. (SER Section 2.1.1.2.3.6.1)

(3) DOE shall provide the NRC staff written notification that the agreements for the six flight restrictions and operational constraints that DOE credits in its frequency analysis (SAR Section 1.6.3.4.1) are in place before commencement of construction to confirm that the technical bases for exclusion of aircraft crash hazards at the GROA from the PCSA that DOE provided in accordance with 10 CFR 63.112(d) remain valid. These restrictions and operational constraints are (i) prohibiting fixed-wing flights below 14,000 ft (mean sea level) within 9 km [5.6 mi] of the North Portal; (ii) 1,000 overflight limit per year above 14,000 ft (mean sea level) within 9 km [5.6 mi] of the North Portal; (iii) overflights are limited to straight and level flights (i.e., maneuvering is not permitted); (iv) carrying ordnance is prohibited within 9 km [5.6 mi] of the North Portal; (v) electronic jamming activities are prohibited within 9 km [5.6 mi] of the

North Portal; and (vi) helicopters are not permitted within 0.8 km [0.5 mi] of facilities that process, stage, or age waste forms. (SER Section 2.1.1.3.3.1.3.3)

(4) DOE shall not, without prior NRC review and approval, take or implement any exception to the IEEE Standards 308–2001, 384–1992, 379–2000, and 603–1998 in the design of the ITS safety interlock subsystems.

Any amendment request must include the design basis for the use of the exception(s), including the ability of structures, systems, and components to perform their intended safety functions assuming the occurrence of event sequences in accordance with 10 CFR 63.112(e)(8). (SER Section 2.1.1.6.3.2.8.2.1)

(5) DOE shall not, without prior NRC review and approval, accept the following waste packages: (i) 5-DHLW/DOE long codisposal; (ii) 2-MCO/2-DHLW codisposal; and (iii) Naval Short.

DOE shall not, without prior NRC review and approval, accept the following canisters: (i) DHLW long; (ii) DOE long; and (iii) Naval Short .

Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of these waste packages and canisters at the GROA or (ii) demonstrates, through the PCSA, that these waste packages and canisters can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112. (SER Section 2.1.1.7.3.9.1)

(6) DOE shall not, without prior NRC review and approval, accept DPCs at the repository.

Any amendment request must include information that either (i) confirms that the current PCSA bounds the intended performance of the DPCs at the GROA or (ii) demonstrates, through the PCSA, that the DPCs can be safely received and handled at the repository during the preclosure period in accordance with 10 CFR 63.112. (SER Section 2.1.1.7.3.9.3.3)

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6.3 Postclosure Safety (NUREG-1949 – SER Volume 3)

As Stated in the SER Volume 3:

> "Volume 3, Repository Safety After Permanent Closure, of this Safety Evaluation Report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the Safety Analysis Report (SAR), Chapter 2: Repository Safety After Permanent Closure, the U.S. Department of Energy (DOE, or the applicant) provided in its June 3, 2008, license application (LA) submittal (DOE, 2008ab), as updated on February 19, 2009 (DOE, 2009av). The NRC staff also reviewed information DOE provided in response to the NRC staff's request for additional information and other information that DOE provided related to the SAR. In particular, this SER Volume 3 documents the results of the NRC staff's evaluation to determine whether the proposed repository design for Yucca Mountain complies with the performance objectives and requirements that apply after the repository is permanently closed. These performance objectives and requirements can be found in NRC's regulations at 10 CFR Part 63, Subparts E and L. The NRC staff's safety evaluation considers the proposed geologic repository's multiple barriers, both natural and engineered (manmade); and the performance assessments (including model abstractions) used for the individual protection, the separate groundwater protection, and the human intrusion evaluations." (SER Volume 3, page xxi)
>
> "Because the DOE's Yucca Mountain performance assessment encompasses such a broad range of technical subjects, the NRC staff used a risk-informed performance-based approach throughout the review process to ensure that the NRC staff's review focused on those items most important to safety and waste isolation. YMRP Section 2.2.1 provides guidance to the NRC staff on how to apply a risk-informed performance-based approach throughout its review of the DOE's Yucca Mountain performance assessment." (SER Volume 3, page xxi)

Key aspects of the review considered multiple barriers, DOE's total system performance assessment (TSPA), and the expert elicitations that support the TSPA. The results of these reviews are summarized in the following passages from SER Volume 3.

> **System Description and Demonstration of Multiple Barriers**
>
> "NRC regulations at 10 CFR Part 63 require that a geologic repository at Yucca Mountain include multiple barriers, both natural and engineered. Barriers prevent or limit the movement of water or radioactive material. A multiple barrier approach ensures that the overall repository system is robust and not wholly dependent on any single barrier. The NRC requires that DOE identify these barriers when it calculates how the repository will perform. DOE is required to describe the capability of each barrier and provide the technical basis for its description. In its SAR for the proposed repository at Yucca Mountain, DOE identified three barriers: the Upper Natural Barrier, the Engineered Barrier System (EBS), and the Lower Natural Barrier. The Upper Natural Barrier is composed of features above the repository (i.e., topography, surficial soils, and the unsaturated zone) that reduce the quantity and rate of movement of water downward toward the repository, which in turn reduces the rate of movement of water from the radioactive waste in the repository to the accessible environment. The EBS includes different engineering features (e.g., emplacement drifts, drip shields, waste packages and its internal components, and emplacement pallets and inverts) that are designed to (i) enhance the performance of the waste package, preventing radionuclide releases while it is intact; (ii) limit radionuclide releases after the waste package is breached by limiting the amount of water that can contact the waste

package; and (iii) limit radionuclide release from the engineered barrier system through sorption processes. The Lower Natural Barrier comprises two features: the unsaturated zone below the repository and the saturated zone, both of which prevent or reduce the rate of radionuclide movement from the repository to the accessible environment through such processes as the slow movement of water and sorption of radionuclides onto mineral surfaces. Each of these barriers include features that DOE described as important to waste isolation. The NRC staff's review of the multiple barriers is provided in SER Section 2.2.1.1.

Review of Postclosure Total System Performance Assessment

DOE conducted an analysis, through its Total System Performance Assessment (TSPA) computer model, that evaluates the behavior of the high-level waste repository due to the potential release of radionuclides from the repository. The performance assessment provides a method to evaluate the range of features (e.g., geologic rock types, waste package materials), events (e.g., earthquakes, igneous activity), and processes (e.g., corrosion of metal waste packages, sorption of radionuclides onto rock surfaces) that are relevant to the behavior of a repository at Yucca Mountain. The NRC staff reviewed the TSPA analytic models and analyses DOE provided in its SAR.

Scenario Analysis and Event Probability

To answer the question, "What can happen?" after the repository is closed, DOE considered a wide range of specific features (e.g., geologic rock types, waste package materials), events (e.g., earthquakes, volcanic activity), and processes (e.g., corrosion of metal waste packages, sorption of radionuclides on rock surfaces) for possible inclusion in (or exclusion from) its Total System Performance Assessment (TSPA) model. Once specific features, events, and processes (FEPs) were selected for inclusion in the TSPA model, DOE then used these FEPs to postulate a range of credible, future scenarios. A scenario is a well-defined sequence of events and processes, which can be interpreted as an outline of one possible future condition of the repository system. Therefore, scenario analysis identifies the possible ways in which the repository environment could evolve so that a representation of the system can be developed to estimate the range of credible potential consequences. After the FEPs are selected and used to postulate scenarios, similar scenarios are grouped into scenario classes, which are screened for use in the TSPA model. The goal of the scenario analysis is to ensure that no important aspect of the potential high-level waste repository is overlooked in the evaluation of its safety.

The NRC staff evaluates the applicant's scenario analysis in four separate SER Sections (2.2.1.2.1.3.1 through 2.2.1.2.1.3.4). Section 2.2.1.2.1.3.1 contains the NRC staff's evaluation of both the applicant's methodology to develop a list of FEPs and DOE's list of the FEPs that it considered for inclusion in the performance assessment. In Section 2.2.1.2.1.3.2, the NRC staff evaluates DOE's screening of its list of FEPs, including DOE's technical bases for the exclusion of FEPs from its performance assessment. DOE's formation of scenario classes and the exclusion of specific scenario classes in DOE's performance assessment are evaluated in Sections 2.2.1.2.1.3.3 and 2.2.1.2.1.3.4, respectively.

The NRC staff's evaluation of the applicant's methodology and conclusions on the probability of events included in the performance assessments is addressed in SER Section 2.2.1.2.2. Hence, SER Section 2.2.1.2.2 is aimed at the second of the three risk questions, "How likely is it to happen?" In SAR Section 2.2.2, DOE identified and described those events that exceeded the probability threshold of 1 chance in 100 million per year (10^{-8} per year) of occurring. The NRC staff's evaluation of the applicant's approach for quantifying the event probabilities and the technical basis for determining the probability estimates assigned to each event type with a probability of 10^{-8} per year or higher are evaluated in SER Section 2.2.1.2.2.

Model Abstraction

The NRC staff's evaluation of the applicant's model abstractions focuses on the consequences of overall repository performance. In particular, the NRC staff's evaluation considers the model abstractions used in DOE's TSPA model to represent the performance (i.e., expected annual doses) of the repository.

The evaluation of the model abstraction process begins with the review of the repository design and the data characterizing the geology and the performance of the design and proceeds through the development of models used in the performance assessment. The model abstraction review process ends with a review of how the abstracted models are implemented in the TSPA model (e.g., parameter ranges and distributions, integration with model abstractions for other parts of the repository system, representation of spatial and temporal scales, and whether the TSPA model appropriately implements the abstracted model). The NRC staff has separated its model abstraction review into 13 categories that are addressed in SER Sections 2.2.1.3.1 through 2.2.1.3.14.

Expert Elicitation

Expert elicitation is a formal, structured, and well-documented process for obtaining the judgments of multiple experts on various scientific topics. Pursuant to 10 CFR 63.21(c)(19), DOE must explain how expert elicitation was used in its application. Consistent with YMRP Section 2.5.4, DOE could elect to use the subjective judgments of experts, or groups of experts, to interpret data and address technical issues and inherent uncertainties when assessing the long-term performance of a geologic repository. In its SAR, the applicant used the results of three formal expert elicitations to complement and supplement other sources of scientific and technical information such as data collection, analyses, and experimentation. The NRC staff has reviewed DOE's use of expert elicitation, which includes a technical review of the results of these elicitations.

SER Section 2.5.4 provides the NRC staff's review of the three expert elicitations DOE used in support of its SAR. Expert elicitations were conducted in the areas of seismic hazard (SAR Section 2.2.2.1); igneous activity (SAR Section 1.1.6.2, Section 2.2.2.2, and Section 2.3.11); and saturated zone flow and transport (SAR Section 2.3.9.2)."
(SER Volume 3, pages xxii to xxiv)

Appendix B (Postclosure Safety Review for a Potential Repository at Yucca Mountain) provides a detailed roadmap of the issues evaluated in each of the Chapters in SER Volume 3. Some of the key concepts of the postclosure review described below are: multiple barriers, deep percolation and seepage into the repository, degradation of the waste package, transport of radionuclides, disruptive events and expert elicitation, and compliance calculations.

6.3.1 Multiple Barriers

It is important to point out that the multiple barrier approach in Part 63 is different from the approach in NRC's regulations at Part 60 for sites other than Yucca Mountain (see Section 3.1.5 and Appendix A for further details). Some stakeholders were concerned when NRC supported a different regulatory approach in Part 63 for multiple barriers than the multiple barrier approach in NRC's generic regulations for sites other than Yucca Mountain (e.g., specific quantitative subsystem requirements, such as a 300-1,000 year waste package lifetime). Specifically, for Part 63, DOE has flexibility in identifying the multiple barriers and is required to support the basis for each barriers capability consistent with the performance assessment used to demonstrate compliance with the individual dose limit. Section 2.1 of this document provides a more comprehensive discussion of basis for NRC adopting a different approach for multiple barriers in Part 63.

As explained in Section 3.3.1 the DOE's SAR and NRC's review, although covering many technical aspects of the repository, was straightforward in its approach. In particular, DOE identified the various components and features of its multiple barriers and explained how such features worked in combination to limit the releases of radionuclides to the compliance point (e.g., upper natural barrier limits the water that can contact waste packages, the engineered barrier system provides containment of the majority of the high-level waste within the waste package thorough out the entire compliance period, and for those radionuclides that exit the waste package, the lower natural barrier reduces the transport of those radionuclides to the accessible environment where there is the potential for human contact). Additionally, the staff's understanding of DOE's multiple barrier approach was used to risk inform its review to ensure those barrier features and components most important to safety were appropriately supported in the models and parameters used in the performance assessment.

It is important to note that the EBS provides a capability, as noted previously, to retain the majority of the inventory in the waste package because the corrosion resistant material remains substantially intact and therefore allows limited water into the package and limited release of radionuclides from inside the package. In particular, (1) releases from most of the waste packages of commercial spent nuclear fuel will be from stress corrosion cracking that only releases radionuclides from the diffusion process and does not allow water to enter the waste package; and (2) general corrosion of the waste packages that would allow water to enter the package and advective releases out of a waste package is of very limited occurrence - DOE predicted that the earliest general corrosion waste package failure (at the 95th percentile) would occur at 560,000 years and at 1 million years. About 10 percent of the waste packages are predicted to fail from general corrosion. In addition, in disruptive scenarios under which appreciable quantities of radionuclides are mobilized, important EBS barrier functions included sorption onto steel corrosion products and attachment to immobile colloids. These processes provided significant sequestration of radionuclides within the waste package (DOE 2009dc). Such significant performance from the EBS may indicate that the repository does not have multiple barriers but relies only on the capabilities of the EBS. While it is true that no releases can occur until the waste package is breached, however, such does not diminish the potential capabilities of the upper natural system and the lower natural barrier as effective barriers. As explained in the SER,

> Lower Natural Barrier (unsaturated zone)
>
> "For sorbing radionuclides, travel times depend on the radionuclide-specific sorption coefficient. More strongly sorbing aqueous species, such as Pu-242, have transport times on the order of hundreds of thousands of years and longer in the southern area. Some radionuclides that are dominant contributors to the total inventory are significantly delayed before reaching the water table due to sorption of radionuclides onto the rock matrix that exists in the southern area (e.g., Cs-137, Sr-90, Pu-239, Pu-240, Am-241, Am-243) [SAR Section 2.3.8]." (SER Vol. 3, page 17-16)

<u>Lower Natural Barrier (saturated zone)</u>

"Flow in the alluvial portion of the flow system is conceptualized as relatively slow because the effective flow porosity is relatively high [average estimated value of 0.18 (SAR p. 2.3.9-59)]. Overall, the transport time for nonsorbing radionuclides ranges from about 10 years to several thousand years (SAR p. 2.3.9-9). Sorbing radionuclides can be significantly delayed by sorption to alluvium mineral grains in which case transport times for strongly sorbing radionuclides generally exceed 10,000 years (SAR p. 2.3.9-9). Table 17-5 provides transport times for select radionuclides representing a range of sorption behavior."
(SER Vol. 3, page 17-17)

Thus, the results of the performance assessment demonstrate, that non-sorbing radionuclides that are more mobile show up in the initial 10,000 years, whereas the sorbing radionuclides are more significant during the longer period after 10,000 years, which extends to 1 million years. However, releases are always within the regulatory dose limits as described in the SER:

"The average annual doses are largest for the seismic ground motion and igneous intrusive modeling cases (generally a factor of 10 or more larger than the other modeling cases; see SAR Figure 2.4-18). Tc-99 (a nonsorbing radionuclide) is the largest contributor to the average annual dose in the initial 10,000 years. Tc-99 accounts for approximately 0.001 mSv/yr [0.1 mrem/yr] of the peak of the overall average annual dose of approximately 0.003 mSv/yr [0.3 mrem/yr] (SAR Figure 2.4-20a). After 10,000 years and up to 1 million years, the peak of the overall average annual dose occurs at 1 million years, with Pu-242 and Np-237 being the largest contributors to the peak of the overall average annual dose. Pu-242 and Np-237 account for approximately 0.01 mSv/yr [1.0 mrem/yr] of the peak of the overall average annual dose of approximately 0.02 mSv/yr [2.0 mrem/yr] at 1 million years (SAR Figure 2.4-20b)."
(SER Vol. 3, page 17-18)

Thus, doses are well within the average annual individual dose limits of 15 mrem (in the initial 10,000 years) and 100 mrem (for the period after 10,000 years up to 1-million years) due to the multiple barriers working in combination to limit releases to the accessible environment.

6.3.2 Deep Percolation and Seepage into the Repository

"In SAR Section 2.1.2.1.6.2, DOE explained that the average percolation flux at the repository depth is, at most, a few percent less than the average net infiltration near the surface above the repository. Because changes in the flow rate of water between the ground surface and the repository level are relatively small, the NRC staff determines that DOE did not attribute barrier capability to any significant processes that result in the diversion of water away from the emplacement drift location. However, DOE explained in SAR Sections 2.1.2.1.2 and 2.1.2.1.6.2 and in DOE (2009an, Enclosures 1 and 2) that capillary diversion of water at the host rock–air interface at the drift wall prevents much of the water flowing in the rock at the repository level from entering the drift as seepage (i.e., dripping). DOE explained that at some drift locations, all of the water is diverted around the drift, resulting in no drips at all; at others, only some of the water enters, and the remainder is diverted around the drift. In addition, the short duration, relatively higher flow rates resulting from infiltration following brief episodes of precipitation are spread out in time and space as they pass through the Paintbrush Tuff. In DOE (2009an, Enclosure 2), DOE explained that this damping of episodic infiltration pulses by the Paintbrush Tuff results in water flow rates below the Paintbrush Tuff that are consistently lower than the peak flow rate during the infiltration pulse, but which are more nearly constant over time (i.e., steady-state fluxes below the Paintbrush Tuff). DOE explained that because capillary diversion processes are more effective at low percolation flow rates, the damping of episodic infiltration pulses by the Paintbrush Tuff contributes to the effectiveness of the capillary barrier. DOE quantified the barrier capability of the unsaturated zone above the repository for each of the five

> percolation subregions for the climate states projected for the first 10,000 years after repository closure (SAR Section 2.1.2.1.2). DOE used an analysis based on the TSPA seepage models and inputs to demonstrate that average seepage rates range from less than 1 to about 17 percent of the percolation fluxes for intact drifts within the first 10,000 years following closure, as described in DOE (2009bo, Enclosure 3, Table 11). DOE expects capillary forces to divert more than 80 percent of percolation flux away from the intact drifts for the initial 10,000 years after closure. DOE (2009bo, Enclosure 3, Table 5) identified that for intact drifts, the fraction of the repository experiencing dripping conditions (i.e., the seepage fraction) ranges from 10 to 70 percent. Results for the collapsed drift case, which is a likely scenario in the post-10,000-year period, show that the mean seepage percentage ranges from about 40 to 56 percent, as described in DOE (2009bo, Enclosure 3, Table 11). DOE expects that capillary forces would divert at least 44 percent of percolation flux away from collapsed drifts. The post-10,000-year seepage fractions for the corresponding flow fields range from about 44 to 89 percent, as described in DOE (2009bo, Enclosure 3, Table 8)." (SER Volume 3, page 1-9)

Table 6-2 displays how the values of precipitation are reduced by processes such as evapotranspiration and runoff that results in a very small amount of water entering repository drifts (e.g., on the order of 2 mm/yr for the initial 10,000 years).

Table 6-2. Quantitative Reduction in Flux from the Ground Surface to Water Entering the Drift Using Flux Averages Over the Repository Footprint (from SER Volume 3, Table 9-1)

<table>
<tr><th rowspan="2"></th><th rowspan="2">Precipitation mm/yr*</th><th rowspan="2">Net Infiltration mm/yr*</th><th rowspan="2">Unsaturated Zone Site-Scale Top Boundary Net Infiltration mm/yr*</th><th rowspan="2">Deep Percolation mm/yr*</th><th colspan="2">Seepage Repository Footprint</th></tr>
<tr><th>Flux mm/yr*</th><th>Fraction of Area</th></tr>
<tr><td>Component of Natural Barrier</td><td>---</td><td>Topography and soils</td><td>---</td><td>Unsaturated Zone</td><td colspan="2">Unsaturated Zone</td></tr>
<tr><td>Primary Feature or Processes</td><td>Semiarid Climate</td><td>Evapo-transpiration, Runoff, Infiltration</td><td>Uncertainty in Net Infiltration</td><td>---</td><td colspan="2">Capillary Diversion and Vapor Barrier</td></tr>
<tr><td>Section of SAR</td><td>2.3.1</td><td>2.3.1</td><td>2.3.2</td><td>2.3.2</td><td colspan="2">2.3.3</td></tr>
<tr><td>Thermal Period[1]</td><td>---</td><td>---</td><td>---</td><td>---</td><td>0</td><td>0</td></tr>
<tr><td>Initial 10,000 years, Nominal[2]</td><td rowspan="2">296.7</td><td rowspan="2">38.88</td><td rowspan="2">21.37</td><td rowspan="2">21.74</td><td>2.0 (6.4)[3]</td><td>0.31</td></tr>
<tr><td>Initial 10,000 years, Seismic[2]</td><td>2.3 (7.4)[3]</td><td>0.31</td></tr>
<tr><td>Post 10,000 years, Nominal</td><td rowspan="2">---</td><td rowspan="2">---</td><td rowspan="2">---</td><td rowspan="2">31.83</td><td>3.4 (8.5)[3]</td><td>0.40</td></tr>
<tr><td>Post 10,000 years, Seismic</td><td>15.5 (22)[3]</td><td>0.69</td></tr>
<tr><td colspan="7">* Units: 25.4 mm/yr = 1 in/yr
1 Thermal period defined by drift wall temperature > 100 °C [212 °F] (SAR Section 2.3.3.3.4).
2 Values of precipitation and percolation for initial 10,000 years are for glacial transition climate.
3 Average flux for seeping environment is in brackets</td></tr>
</table>

The NRC also evaluated DOE’s consideration of uncertainty in the net infiltration. For example:

"The applicant's analyses consistently identified uncertainty in net infiltration, which is closely related to percolation fluxes within the unsaturated zone, as significantly affecting uncertainty in intermediate results (e.g., drift seepage, drift wall temperatures, radionuclide releases from the EBS, unsaturated zone radionuclide transport rates) and expected mean annual doses (SNL, 2008ag). DOE's analyses also identified (i) relatively smaller contributions to uncertainty in seepage into drifts arising from uncertainty in host rock permeability and capillary strength and (ii) contributions to uncertainty in in-drift temperature and relative humidity from host rock thermal conductivity, as outlined in SNL (2008ag, Section K4).

The NRC staff examined DOE's sensitivity analyses with respect to intermediate results and expected mean annual dose by comparing the sensitivity results with the DOE description of the physical processes governing barrier function as represented in the models used for performance assessment. The NRC staff's review focused on parameters that DOE identified as systematically affecting either intermediate results or expected mean doses. On the basis of the NRC staff comparison, the NRC staff concludes that DOE's sensitivity analysis is consistent with the DOE description of the physical processes embodied in models used for performance assessment calculations with respect to net infiltration, percolation fluxes, thermal responses in the host rock, seepage, and in-drift temperature and humidity.

The NRC staff's conclusions on DOE's procedures for propagating uncertainty in performance assessment consider the reasonableness of overall performance assessment results. This is based in part on the NRC staff review of the performance assessment calculations (see SER Section 2.2.1.4.1). DOE's rankings of key parameter inputs are consistent with its representation of engineered barrier characteristics and are derived from a variety of approaches for sensitivity analyses, as described in SNL (2008ag, Appendix K9). The ranking suggest that failure and release mechanisms are more important to repository performance than natural system factors of the unsaturated zone because of the longevity of the engineered barriers. The NRC staff found DOE rankings of key parameter inputs to be similar to the NRC staff's parameter rankings derived from uncertainty analyses performed using independent methods and models (NRC, 2005aa)." (SER Volume 3, page 9-10)

The NRC staff's evaluation of DOE's abstraction of water flow into the drifts, including any thermal-hydrologic effects, was summarized as follows.

"The NRC staff has reviewed the information DOE provided and concludes that the performance and treatment of the uncertainty for the seepage rate and fraction are reasonable and acceptable because they are consistent with the technical justification provided for the model abstractions and the barrier capabilities. In evaluating seepage rates for performance assessment, the NRC staff notes that the 10,000-year and million-year periods are considered separately. In particular

- For the initial 10,000 years after disposal, mean values of seepage fraction and rate and their uncertainty are not important for performance assessment, because drip shields are predicted to remain intact well beyond 10,000 years. Intact drip shields divert seeping water away from waste packages.

- The thermal seepage abstraction that shows no seepage occurring when the drift wall temperature exceeds 100 °C [212 °F] is acceptable because drip shields remain intact well beyond the thermal period and therefore divert any seeping water away from waste packages.

- For the period from 10,000 years to 1 million years, DOE's seepage and dose calculations are dominated by the seismic ground motion seepage scenario. Average seepage fraction (69 percent) and rate (49 percent of percolation) are acceptable because reasonable increases accounting for uncertainty would not significantly affect performance assessment calculations and dose results.

- Seepage estimates for the igneous intrusion and seismic fault displacement modeling cases are acceptable because DOE uses conservative assumptions for estimating seepage." (SER Volume 3, pages 9-41 and 9-42)

6.3.3 Degradation of the Drip Shield and Waste Package

Degradation of the drip shield and waste package considers both the chemical degradation (i.e., corrosion) and the mechanical disruption (e.g., due to seismic events and rockfall loads). DOE discussed the significance of the capability of the drip shield to provide protection from seepage water dripping onto the waste package over the initial 12,000 years during which the waste packages are sufficiently hot that specific chemical and thermal conditions may support localized corrosion on the surface of the waste package.

"DOE calculated that conditions in the drift (e.g., temperature, pH, seepage water chemistry) may support localized corrosion of the waste package if the drip shield fails and allows seepage water to contact the waste package within approximately 12,000 years after repository closure, as detailed in DOE (2009dg, Enclosure 1). Following 12,000 years after repository closure, DOE calculated that there is a low probability for aqueous chemical conditions in the drift to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package, as shown in DOE (2009dg, Enclosure 1, Figure 1). The TSPA analysis calculates that few drip shields will fail within 12,000 years after repository closure. Therefore, the probability of waste package breach by localized corrosion is low in DOE's model." (SER volume 3, page 4-4).

6.3.3.1 Chemical Degradation

"DOE's models for chemical degradation of the engineered barrier systems focus on both the drip shield and the waste package outer barrier. Consistent with the YMRP guidance, the NRC staff performed a risk-informed, performance-based review, focusing on those aspects of DOE' s models for chemical degradation of the drip shield and the waste packages that are most important to the assessment of barrier capability. DOE concluded that seepage flux is the primary source of water that may react with the engineered barrier system components (SAR Section 2.3.7.12.1). In DOE's model for flow of seepage water through the engineered barrier system, the water must first pass through the drip shield and then through the waste package before contacting and mobilizing the waste form. As such, SER Section 2.2.1.3.1 first concentrates on DOE's models for chemical degradation of the drip shield and then addresses DOE's models for chemical degradation of the waste package." (SER Volume 3, page 4-3)

Drip Shield

"The drip shield, which DOE described in SAR Section 1.3.4.7, is an engineered metal barrier designed to divert water that enters the drift and prevent it from contacting the waste package. DOE stated that the drip shield will be fabricated from Titanium Grade 7 (UNS R52400). Titanium Grade 7 is a commercially pure titanium alloy with the addition of a small amount of palladium (approximately 0.2 weight percent) to enhance its corrosion resistance. The drip shield structural supports will be fabricated from Titanium Grade 29 (UNS R56404), which is a titanium alloy composed of approximately 6 weight percent aluminum and 4 weight percent vanadium for strength, plus approximately 0.1 weight percent ruthenium for corrosion resistance.

In developing the postclosure performance assessment analysis, DOE evaluated a number of FEPs (in SAR Table 2.2-5) related to chemical degradation of the drip shield, including

- General corrosion of the drip shields (FEP 2.1.03.01.0B)
- Stress corrosion cracking of the drip shields (FEP 2.1.03.02.0B)
- Localized corrosion of the drip shields (FEP 2.1.03.03.0B)
- Hydride cracking of the drip shields (FEP 2.1.03.04.0B)
- Microbially influenced corrosion (MIC) of the drip shields (FEP 2.1.03.05.0B)
- Early failure of the drip shields (FEP 2.1.03.08.0B)
- Oxygen embrittlement of the drip shields (FEP 2.1.06.06.0B)
- Creep of metallic materials in the drip shield (FEP 2.1.07.05.0B)
- Localized corrosion on drip shield surfaces due to deliquescence (FEP 2.1.09.28.0B)
- Thermal sensitization of the drip shields (FEP 2.1.11.06.0B)

With the exception of general corrosion and early failure of drip shields, these features, events, and processes were screened out from the performance assessment on the basis of low consequence or low probability (SAR Table 2.2-5). The NRC staff's evaluation of DOE's bases for excluding these FEPs from the performance assessment is addressed in SER Section 2.2.1.2.1.

With respect to the FEPs included in the performance assessment, DOE described general corrosion of the drip shield as the uniform thinning of both the Titanium Grade 7 drip shield plates and the Titanium Grade 29 structural supports (SAR Section 2.3.6.8.1.1). In SAR Section 2.2.2.3, DOE defined drip shield early failure as through-wall penetration caused by manufacturing- and handling-induced defects at a time earlier than would be expected for a nondefective drip shield." (SER Volume 3, pages 4-3 to 4-4)

The passivity of the drip shield material was a key aspect of the review in supporting its resistance to corrosion processes.

"The NRC staff reviewed DOE's assessment of drip shield passivity. On the basis of its review of the information DOE provided in BSC (2004as, Section 6.5.7), the NRC staff finds that uncertainty in the long-term persistence of the titanium passive film is primarily related to potential passive film degradation by fluoride-bearing brines. The NRC staff concludes that there is no evidence of localized corrosion of Titanium Grade 7 exposed to fluoride-bearing simulated concentrated water for 5 years and thus that the passive film is stable when in contact with a brine having this composition. The NRC staff finds that, based on NRC staff's independent tests, the extent of titanium passive film degradation will generally decrease with decreasing fluoride concentration in the brines (Brossia, et al., 2001aa)." (SER Volume 3, page 4-6)

"In addition, the NRC staff reviewed DOE's assumption that the general corrosion rate is independent of temperature. The NRC staff concludes that the studies DOE cited (e.g., Smailos and Köster, 1987aa; Hua and Gordon, 2004aa) considered materials analogous to the drip shield titanium alloys and environmental conditions that are similar to or more aggressive than those expected in the repository. On the basis of information provided in these studies, the NRC staff concludes that, although the corrosion rates of titanium alloys may have some temperature dependence during an initial period of exposure to corrosive brines, there is little temperature dependence after this period. The NRC staff finds that the information DOE provided is also consistent with independent analyses by NRC (Mintz and He, 2009aa), which confirmed that corrosion rates for titanium alloys do not show significant temperature dependence for temperatures that are representative of the repository conditions. Therefore, the NRC staff finds DOE's assumption that the corrosion rates of the drip shield titanium alloys are independent of temperature acceptable because this assumption will not underestimate the corrosion rate." (SER Volume 3, page 4-7)

"The NRC staff reviewed DOE's model abstraction for general corrosion of the drip shield that was implemented in the TSPA code. On the basis of its review, the NRC staff finds that DOE used appropriate experimental tests and other independent technical literature to provide adequate support for the model abstraction. In addition, DOE appropriately identified and adequately considered processes and features such as the general corrosion of the drip shield that affect barrier capabilities for the initial 10,000 year period, and projected these processes and features beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds that DOE acceptably accounted for general corrosion of the drip shield in the TSPA model." (SER Volume 3, page 4-14)

The NRC staff also considered the potential for early failure of the drip shield, stating:

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"The NRC staff reviewed DOE's model abstraction for early failure of the drip shield that was implemented in the TSPA code. On the basis of its review, the NRC staff finds that DOE provided adequate support for its model abstraction. In addition, DOE appropriately identified and adequately considered features and events such as drip shield early failure that affect barrier capabilities for the initial 10,000 year period, and projected these features and events beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds that DOE acceptably accounted for drip shield early failure in the TSPA model." (SER Volume 3, page 4-16)

Waste Package

"In developing the postclosure performance assessment, DOE evaluated numerous features, events, and processes in SAR Table 2.2-5 related to chemical degradation of the waste package. These FEPs include

- General corrosion of waste packages (FEP 2.1.03.01.0A)
- Stress corrosion cracking of waste packages (FEP 2.1.03.02.0A)
- Localized corrosion of waste packages (FEP 2.1.03.03.0A)
- Hydride cracking of waste packages (FEP 2.1.03.04.0A)
- Microbially influenced corrosion (MIC) of waste packages (FEP 2.1.03.05.0A)
- Internal corrosion of waste packages prior to breach (FEP 2.1.03.06.0A)
- Early failure of waste packages (FEP 2.1.03.08.0A)
- Creep of metallic materials in waste packages (FEP 2.1.07.05.0A)
- Localized corrosion on waste packages' outer surface due to deliquescence (FEP 2.1.09.28.0A)
- Thermal sensitization of waste packages (FEP 2.1.11.06.0A)

DOE included general corrosion, stress corrosion cracking, localized corrosion, MIC, and early failure in the postclosure performance assessment. Other FEPs were screened from the performance assessment on the basis of low consequence or low probability (SAR Table 2.2-5). The NRC staff's evaluation of DOE's bases for excluding these FEPs from the performance assessment is found in SER Section 2.2.1.2.1.3. (SER Volume 3, page 4-17)

"In SAR Section 2.3.6.3.1 and SNL (2007al), DOE indicated that the stability of the Alloy 22 passive film depends primarily upon its physical and chemical properties, including microstructure, composition, and thickness. On Alloy 22 corrosion specimens, DOE investigated these passive film properties with various surface analytic techniques, including Auger electron spectroscopy, transmission electron microscopy, x-ray photoelectron spectroscopy, and electron energy loss spectroscopy (Orme, 2005aa). DOE performed short-term polarization tests, exposing Alloy 22 samples at 90 °C [194 °F] to solutions with a range of chemical compositions that DOE assumed were similar to, or more aggressive than, those expected in the repository (Orme, 2005aa). The solutions used in short-term polarization tests were either buffered 1 M NaCl solutions or multi-ionic solutions, including simulated acidified water, simulated concentrated water, and basic saturated water (compositions given in SAR Table 2.3.6-1). To assess the long-term passive film behavior, DOE examined 5-year U-bend samples of Alloy 22 exposed to simulated acidified water, simulated concentrated water, and simulated dilute water at 90 °C [194 °F] (Orme, 2005aa)."
(SER Volume 3, page 4-18)

"The NRC staff reviewed DOE's approach to establish the stability of the waste package passive film in repository conditions. The NRC staff determines that DOE's assumption that the passive film will remain stable during the postclosure period is based, in part, on tests of Alloy 22 specimens in a range of corrosion test solutions, including simulated acidified water, simulated concentrated water, simulated dilute water, and basic simulated water (NRC staff's evaluation of FEP 2.1.09.28.0A, SER Section 2.2.1.2.1.3.2). The NRC staff concludes that stability of the waste package passive film may depend on such factors as the pH and concentration of ionic species in water that contacts the waste package. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate in assessing waste package passive film stability in repository conditions. The NRC staff finds that the corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in DOE's near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in NRC staff's independent analysis of in-drift water evolution, described in SER Section 2.2.1.3.3.3.2. Therefore, the NRC staff concludes that DOE's observations of Alloy 22 passive film stability in simulated acidified water, simulated concentrated water, simulated dilute water, and basic simulated water support DOE's assumption of waste package passive film stability in repository conditions because the simulated water chemistries bound repository conditions. This conclusion is supported by similar descriptions of the passive film for Alloy 22 and analogous nickel-based alloys (e.g., Alloy C-4, C276, 600, 625, and 690) reported by NRC staff (Dunn, et al., 2005aa) and others (Lloyd, et al., 2003aa, 2004aa; Gray, et al., 2006aa; Montemor, et al., 2003aa; Hur and Park, 2006aa; Mintz and Devine, 2004aa)." (SER Volume 3, page 4-19)

The NRC staff also considered the potential for localized corrosion, stress corrosion cracking, and early failure of the waste package.

"The NRC staff reviewed DOE's models for localized corrosion of the waste package outer barrier that were implemented in the TSPA code. On the basis of its review, the NRC staff finds that DOE used appropriate experimental tests and applicable technical literature to provide adequate support for the localized corrosion initiation and propagation models. For the first 12,000 years after repository closure, the NRC staff concluded that waste package breach by localized corrosion is unlikely because the intact drip shields will prevent seepage waters from contacting the waste package (SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6). For the time period beyond 12,000 years after repository closure, the NRC staff concludes that DOE's models showed a low probability for localized corrosion initiation because the proposed repository environment (i.e., temperature, pH, and chemical composition of seepage waters) will not support the initiation of localized corrosion. In addition, DOE appropriately identified and adequately considered features and processes such as corrosion potential, relative humidity, localized corrosion initiation and propagation rate, temperature, pH, and chemical composition of seepage waters, that affect the waste package outer barrier capabilities for the initial 10,000 year period, and projected these features and processes beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds acceptable DOE's analytic models for localized corrosion of the waste package outer barrier in the TSPA model." (SER Volume 3, page 4-42)

"The NRC staff reviewed DOE's model abstraction for stress corrosion cracking of the waste package outer barrier that was implemented in the TSPA code. The NRC staff finds that DOE used appropriate experimental tests and other independent technical literature to provide adequate support for its model abstraction. In addition, DOE appropriately identified and adequately considered features and processes such as stress corrosion crack initiation and propagation, crack opening area, and stress, and events such as seismic ground motion, that affect the waste package outer barrier capabilities for the initial 10,000 year period, and projected these features, events and processes beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds that DOE acceptably accounted for stress corrosion cracking of the waste package outer barrier in the TSPA model."
(SER Volume 3, page 4-55)

"The NRC staff reviewed DOE's model abstraction for early failure of the waste package outer barrier that was implemented in the TSPA code. The NRC staff finds that DOE provided adequate support for the model abstraction. In addition, DOE appropriately identified and adequately considered features and events such as waste package type, waste package early failure probability, that affect the waste package

outer barrier capabilities for the initial 10,000 year period, and projected these features and events beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds DOE's accounting for waste package early failure in the TSPA model acceptable."
(SER Volume 3, page 4-57)

6.3.3.2 Mechanical Disruption

"According to DOE, the functions of the drip shield are to prevent rocks from falling on the waste packages and to prevent water from contacting the waste package surface after emplacement when waste packages are still hot, thereby minimizing the potential for corrosion. The purpose of the waste package is to protect the waste form and isolate the radionuclides or slow down their rate of release to the accessible environment. To estimate the effects on timing and magnitude of radionuclide release, DOE analyses considered potential loads from seismic events and the resulting mechanical disruption of the EBS components. DOE considered gradual drift degradation due to thermal loads, time-dependent weakening, and seismic events as sources of generating loads from rubble accumulation on and around the drip shields.

However, DOE excluded the effects of drift degradation due to thermal loads and time-dependent weakening from its Total System Performance Assessment (TSPA) code. The NRC staff's review of DOE's technical bases for exclusion of features, events, and processes (FEPs) (FEP 2.1.07.02.0A, Drift Collapse) is presented in SER Section 2.2.1.2.1.3.2. The scope of this SER Section is limited to reviewing how DOE considered the effects of seismic disruption (i.e., vibratory ground motion and fault displacement) and used the results in the performance analysis." (SER Volume 3, pages 5-1 and 5-2)

"The NRC staff evaluated DOE's assessment of potential degradation of emplacement drifts due to seismic events and estimates of drip shield loading resulting from rubble accumulation and finds that

- DOE adequately considered in the process level models features such as lithophysae and rock fractures, processes such as thermal loading, and time dependent weakening of rocks and events such as seismic, in analyzing the degree of drift degradation during the initial 10,000 year period, and projected these features, events, and processes beyond the 10,000 year post-disposal period through the period of geologic stability.
- DOE used alternative conceptual models including empirical, analytical, and numerical models and acceptable methodologies for estimating the timing and extent of drift degradation due to seismic events including appropriate consideration of thermal loads and time-dependent weakening of excavated drifts that may be potentially subject to repetitive seismic events.
- DOE appropriately considered relevant geologic data obtained during site characterization and the variability of mechanical properties of rocks and associated uncertainties in parameters used in the models in the supporting analyses.
- DOE justified the abstraction of rock rubble loads due to seismic degradation of excavated drifts and the appropriate use of associated parameters for TSPA inputs. (SER Volume 3, pages 5-20 and 5-21)

Drip Shield

"Based on its review of the seismic hazard presented by DOE, the NRC staff determines that complete drift collapse from seismic events could occur during the first 12,000 years of repository closure although such a scenario would be unlikely. The DOE information did not adequately address the uncertainties associated with rockfall load and temperature effects for a potential scenario corresponding to this low probability. In response to an NRC staff RAI, DOE provided additional information in DOE (2009bp, Enclosure 7), which assessed the potential temperature effects at 120 °C [248 °F] at 10 percent rockfall loads. Using the methods DOE developed to evaluate 10 percent of rockfall loads, the NRC staff extended this approach to

100 percent of potential rockfall loads. Using the information in SAR Table 2.3.4-43, the NRC staff finds that an approximately 30 percent reduction in drip shield plate thickness minimally increases the likelihood of plate rupture from approximately 1 percent to approximately 5 percent. Although some strength properties show 30 percent variations from 60 to 120 °C [140 to 248 °F], the NRC staff concludes that these variations are expected to have only a small to negligible effect on the likelihood of plate rupture for 100 percent collapsed drifts. This is because the titanium plate will have increased ductility and, thus, increased its ability to accommodate deformation without rupture under loads associated with unlikely seismic events. In addition, the NRC staff considers the loads that could potentially increase the likelihood of plate rupture are associated only with earthquakes having $<5 \times 10^{-7}$ annual likelihoods. Using insights from the TSPA model, the NRC staff concludes that potential changes in the likelihood of plate rupture on the order of several percent would not affect the performance assessment significantly. The NRC staff concludes that DOE's sensitivity analyses, as presented in DOE (2010ac), also demonstrate that potential changes in the likelihood of plate rupture on the order of several percent would not affect the performance assessment significantly. Therefore, the NRC staff concludes that DOE's use of titanium alloy material properties at 60 °C [140 °F] is an acceptable approach for evaluating postclosure repository performance because uncertainties associated with potentially higher temperatures would not significantly affect the results of the performance assessment." (SER Volume 3, pages 5-24 and 25)

"DOE relies on the drip shields as effective barriers to advective water flow or rock rubble impacts on the waste package. The NRC staff reviewed the information DOE presented relevant to the barrier capability of the drip shield and finds the following:

- DOE appropriately identified and adequately considered potential events and processes such as fault displacement and seismically induced drift collapse that may lead to openings in the drip shield that affect barrier capabilities for the initial 10,000 year period, and projected these events and processes beyond the 10,000 year post-disposal period through the period of geologic stability.

- DOE used acceptable models and information to demonstrate that potential openings from horizontal or vertical displacements during seismic events would not affect performance significantly.

- DOE acceptably assumed that fault displacements sufficient to damage a waste package remove all barrier capabilities from the associated drip shield.

- DOE appropriately evaluated the potential for ruptures in the drip shield plates during the first 12,000 years of closure by taking a conservative approach.
- DOE appropriately determined that a small likelihood exists for such ruptures if earthquakes with annual probabilities of exceedance of $<5 \times 10^{-7}$ occur. DOE adequately implemented this likelihood of plate failure in the TSPA.

- DOE adequately demonstrated that uncertainties in this information would not affect the results of the performance assessment significantly.

The NRC staff finds that DOE adequately evaluated the barrier capabilities of the drip shield mechanical disruption due to seismic events and has appropriately incorporated the risk-significant aspects of this evaluation into the performance assessment calculations." (SER Volume 3, page 5-29)

Waste Package

"Information presented in SAR Table 2.1-3 suggests that seismic ground motion damage to the EBS components is an important mechanism that affects the EBS capability to perform its intended functions. DOE stated in DOE (2009bl, Enclosure 1) that seismically-induced waste package damage is more significant in early times and that nominal failure processes are more significant at later times. According to DOE, seismically-induced stress corrosion cracking is the most probable waste package damage mechanism. The majority of commercial spent nuclear fuel (CSNF) and CDSP waste package failures due to seismically induced stress corrosion cracking occur prior to drip shield plate/crown failure, as described in DOE (2009bl, Enclosure 1, Figures 5 and 6).

As described in the following list, DOE considered three idealized states of the EBS (SAR Section 2.3.4.5):

1. Structurally stable drip shield state (intact drip shield)—when the waste packages are free to move and may be damaged due to impacts with other components of the EBS during seismic events

2. Drip shield framework failure state (collapsed drip shield)—when the drip shield–waste package interactions during seismic events may damage the waste package outer barrier

3. Drip shield plates failure state—when the waste package is surrounded by and in direct contact with rubble and may be damaged due to waste package–rubble interactions during seismic events

As DOE detailed in DOE (2009bl, Enclosure 1, Figure 1), nominal stress corrosion cracking in a CSNF waste package would initiate between 200,000 and 300,000 years, when the timeframe is dependent on the drip shield performance (reviewed in SER Section 2.2.1.3.2.6). The initiation of stress corrosion cracking would occur after the beginning of Idealized State 2. The CSNF waste packages cannot move as freely in Idealized State 2 as in Idealized State 1, thereby reducing the potential for seismically induced stress corrosion cracking.

For the three idealized states, DOE considered two waste package failure modes.

1. The first failure mode is referred to as ‘the residual stress failure mode’ in this SER section. The waste package damage is expressed in terms of the waste package outer corrosion barrier surface area that may be susceptible to stress corrosion cracking. It is defined as an area with the residual stresses exceeding one of three residual stress threshold values: 90, 100, and 105 percent of the Alloy 22 yield stress (reviewed in SER Section 2.2.1.3.1.3.2.3).

2 The second failure mode is referred to as ‘the tensile tearing failure mode’ in this SER section. DOE used Alloy 22 ultimate tensile strain as a failure criterion to evaluate the waste package outer barrier tensile tearing (rupture and/or puncture) occurrence.

For these two failure modes, DOE developed the abstractions using a three-part approach: (i) the rupture/puncture probability was defined as a function of PGV and the effective tensile stress limits, (ii) the probability of a nonzero damaged area was defined as a function of PGV and the residual stress threshold damage, and (iii) for nonzero damaged area cases, a conditional probability distribution for the magnitude of the conditional damaged area was defined as a function of PGV and the residual stress threshold.

DOE’s analyses results indicate greater mechanical damage potential to the waste package during Idealized State 1. However, the NRC staff reviewed the fundamental aspects of damages in all three idealized states and their abstractions.” (SER Volume 3, pages 5-30 and 5-31)

The NRC staff review concluded “that the technical bases for TSPA waste package abstractions presented in the SAR are adequately supported because

- DOE adequately considered processes and events such as outer corrosion barrier thinning and mechanical damage due to collisions during seismic events, in analyzing the degree of mechanical disruption of waste packages during the initial 10,000 year period, and projected these processes and events beyond the 10,000 year post-disposal period through the period of geologic stability.

- DOE adequately addressed uncertainties in the waste package conditions and the environmental effects on the waste package components

- For characterizing waste package damage, DOE followed established practice for mechanical and structural performance assessment

- DOE used appropriate seismic loading conditions that are consistent with the values presented on the bounded hazard curve

- To evaluate the waste package damage, DOE used failure criteria that are consistent with accepted industry practice and/or widely used criteria in the field of mechanical and structural engineering

- For calculating the residual stress and establishing tensile tearing failure modes, DOE used analytical and numerical methods that are appropriate for the types of analyses

In summary, the NRC staff finds acceptable DOE's technical bases for the waste package abstractions it used and that it adequately represented waste package performance in the TSPA abstractions."
(SER Volume 3, page 5-47)

6.3.4 Radionuclide Release Rates

The NRC staff's review of the TSPA abstraction of in-package and below-package conditions and processes affecting release of radionuclides to the natural system is summarized in the following passages from SER Volume 3.

"The engineered barrier system and the transport pathway within the drift (repository tunnel) are the initial barriers to aqueous radionuclide release. If a waste package is breached and water enters the waste package, the radionuclides contained in the package may be transported from the engineered barrier system. The processes that could lead to this radionuclide release are affected by the chemical characteristics of the water, which in turn are affected by the materials that interact with the water. Therefore, as required by 10 CFR 63.113 and 10 CFR 63.114, the performance assessment analysis models radionuclide release rates from the engineered barrier system because these rates would significantly affect the timing and magnitude of transport for any radionuclide released from the repository.

The applicant identified five models it considered important for abstracting radionuclide releases from the engineered barrier system. The five models the applicant identified and the associated sections in this SER Section that address them are

1. The in-package chemical and physical environment model (SER Section 2.2.1.3.4.3.1) used to establish the conditions under which waste forms degrade and radionuclides are mobilized

2. The waste form degradation model (SER Section 2.2.1.3.4.3.2) used to calculate the rate at which the waste form degrades and the radionuclides become available for release

3. The concentration limits model (SER Section 2.2.1.3.4.3.3) used to apply chemically based upper limits on dissolved concentrations of some radionuclides

4. The availability and effectiveness of colloids model (SER Section 2.2.1.3.4.3.4) used to calculate the stabilities and concentrations of various types of colloids (small suspended particles that may mobilize radionuclides in water)

5. The engineered barrier system radionuclide transport model (SER Section 2.2.1.3.4.3.5) used to simulate radionuclide transport from the waste form, through the waste package, and out of the engineered barrier system

The FEPs that DOE identified as relevant to radionuclide release rates and solubility limits are listed in the applicant's SAR Section 2.3.7.2 and Table 2.3.7-1. The NRC staff evaluates the rationales for excluding FEPs from the performance assessment model in SER Section 2.2.1.2.1.3.2. In this SER section, the NRC staff finds acceptable the applicant's bases for the list of FEPs considered and excluded from the TSPA code analysis that are relevant to waste form behavior, solubility limits, colloidal transport, and radionuclide release rates." (SER Volume 3, pages 7-1 and 7-2)

In-Package Chemistry

"The NRC staff evaluated the chemistry of incoming waters used in the in-package chemistry model to simulate seepage water input. The NRC staff independently verified that the chemical compositions of the pore waters and the J–13 well water span the range of predominant water types found in the Topopah Spring welded tuff at Yucca Mountain, as described in SNL (2007ak, Section 6.6.5). The NRC staff finds that the applicant adequately limited uncertainty in the initial chemistry of water entering the waste package by incorporating a range of Yucca Mountain pore water and basalt water chemistries in developing the in-package chemistry model abstractions. In BSC [2005ad, Sections 6.5(a) and 6.6(a)] and in SAR Figures 2.3.7-13 through 2.3.7-18, the applicant demonstrated limited sensitivity of in-package chemistry to the incoming water composition. Therefore, the NRC staff finds that the chemistry of incoming waters the applicant used to simulate seepage water input are acceptable for developing the in-package chemistry model abstractions." (SER Volume 3, page 7-7)

The NRC staff evaluated the modeling approach and information the applicant used to generate and support the in-package chemistry abstractions for pH. The NRC staff finds that the solubility limit approach used to quantify lower and upper pH limits is consistent with accepted geochemical principles. The NRC staff evaluated information provided in DOE (2009ax, Enclosure 4) and finds that the choice of waste package design the model used (e.g., a 5-DHLW/DOE codisposal waste package containing five high-level waste glass containers versus the 2-MCO/2-DHLW codisposal waste package containing two high-level waste glass canisters) does not affect established pH limits because the pH limits are based on buffering reactions that are not influenced by the total volumes and surface areas of material components in the waste form cells. In addition, on the basis of its review of information provided in BSC [2005ad, Sections 6.5(a), 6.6.3(a), and 6.6.5(a)], the NRC staff also finds that lower and upper pH limits defined for each waste form cell in the pH abstractions are appropriate because they are within the pH trends observed in time-dependent basecase EQ6 simulations at different incoming water chemistries and in sensitivity analyses at varying $p\mathrm{CO_2}$ values and material degradation rates. On the basis of open literature reviews, the NRC staff finds that the phases the applicant predicted to form and control pH in the waste form cells (i.e., trevorite and schoepite) are consistent with the phases reported as alteration products in steel corrosion and $\mathrm{UO_2}$ degradation experiments, as well as in phases observed at natural analogs (Wang, et al., 2001aa; Da Cunha Belo, et al., 1998aa; BSC, 2004ah; Wronkiewicz, et al., 1996aa; Langmuir, 1997aa; Pearcy, et al., 1994aa)." (SER Volume 3, page 7-10)

Waste Form Degradation

"Commercial SNF is composed of irradiated fuel rods from PWRs and boiling water reactors. High-level waste glass is made by melting high-level radioactive materials with silica and/or other glass-forming chemicals and then solidifying them. DOE SNF (including naval SNF) comes from a range of high-level waste generators, from noncommercial reactors, and from the use of radioactive material that encompasses a variety of fuel types. On the basis of the significance to risk, the NRC staff's review focused on the inventory of radionuclides and radionuclide distribution in the following areas: commercial SNF; degradation of commercial SNF; degradation of high-level waste glass; degradation of DOE SNF, naval SNF, and cladding; and associated model and data uncertainties, including waste form degradation under disruptive scenarios and microbial effects. Each waste form has a specific radionuclide inventory. In the nominal scenario, the waste form degrades as it dissolves after the cladding, if any, corrodes and fails in the aqueous environment. In seismic or igneous scenarios, mechanically or thermally assisted degradation could also occur. For the applicant's waste form degradation abstractions in the TSPA code, the input information includes the design description of the waste package, the waste form, the waste package internals, and in-package water chemistry and temperature. The output from this section includes waste form mobilization rates to assess engineered barrier system radionuclide transport."
(SER Volume 3, page 7-13)

"The NRC staff evaluated the applicant's processes and modeling for matrix dissolution and radionuclide release from the gap and grain boundaries. The NRC staff reviewed open literature information for this evaluation, including results from the NRC staff's independent modeling (NRC, 2008aa; Leslie, et al., 2007aa; Jain, et al., 2004aa). The NRC staff finds acceptable the applicant's conclusion that the release of high-solubility radionuclides (e.g., I–129 and Tc–99) will be at the same rate of oxidative UO_2 matrix dissolution, whereas release of low-solubility radionuclides (e.g., plutonium isotopes) may be limited by solubility. These findings are consistent with laboratory test results (Wilson and Gray, 1990aa).

The UO_2 matrix would dissolve in the oxidizing environment expected at the proposed Yucca Mountain repository (Shoesmith, 2000aa). This is consistent with the alteration process of natural analog uraninite (BSC, 2004ah). The applicant's mathematical models are empirical. The applicant identified important environmental and commercial SNF parameters controlling the dissolution rate. Those parameters included oxygen partial pressure, carbonate concentration, temperature, pH, and the surface area of the matrix contacted by water. These parameter values were obtained from accelerated test results in oxidizing environments, with consideration of data uncertainties. The NRC staff finds these parameters and their values acceptable because they were derived from appropriate tests that are based on electrochemical, flow-through, and round-robin tests. In addition, the NRC staff performed independent evaluations (Leslie, et al., 2007aa) that confirmed the applicant's conclusions. The NRC staff also finds acceptable the applicant's assumption that radionuclide release from the gap and grain boundaries is rapid because these radionuclides are not atomically bound in the matrix. Therefore, the NRC staff finds that the applicant conservatively presented the conceptual and mathematical models for radionuclide releases that are faster than those expected to occur from matrix degradation and from the gap and grain boundaries."
(SER Volume 3, page 7-16)

"The NRC staff finds acceptable the model abstraction of the degradation of high-level waste glass because it incorporates the applicant's waste form design features, which include thermal loading, structural characteristics, radionuclide inventory, chemical composition, and microstructural characteristics (SAR Section 1.5.1). The NRC staff evaluated the processes and modeling that the applicant presented for the dissolution of high-level waste glass. The NRC staff's evaluation was based on open literature information originally compiled by Leslie, et al. (2007aa)." (SER Volume 3, page 7-18)

"The NRC staff finds acceptable the applicant's assumption that DOE's SNF will degrade instantaneously because it would not underestimate the radiological consequences. The applicant's approach to model the naval SNF as commercial SNF is also acceptable because this modeling assumption would not underestimate the radiological consequences. The naval SNF is more robust and would release less radionuclides (BSC, 2004ao)." (SER Volume 3, page 7-20)

"The NRC staff reviewed the degradation models implemented in the TSPA code abstraction to confirm that they provide consistent results with the output from the detailed process-level models and/or empirical observations on the characteristics of commercial SNF and high-level waste glass, as described in this SER section. The applicant presented a bounding assumption regarding radionuclide release from all waste forms under igneous intrusive conditions in SAR Section 2.3.11.3.2.4. The applicant assumed that all waste forms instantaneously degrade to be mobilized for release. The NRC staff finds that these instantaneous degradation models are conservative and bounding in terms of soluble radionuclide release. As discussed earlier, in the TSPA model the release from the failed waste package is insensitive to the fast rate of waste form degradation. The release rate is controlled mainly by the inventory of each waste package and waste package failure rate for high-solubility radionuclides and concentration limits of dissolved species or colloids for low-solubility radionuclides." (SER Volume 3, page 7-21)

Solubility Limits

"The applicant's abstraction for concentration limits calculates concentration limits for plutonium, neptunium, uranium, thorium, americium, tin, and protactinium using lookup tables (SNL, 2007ah) that define values (in mg/L, a unit that is approximately equivalent to parts per million, or ppm) as functions of pH and fCO_2 (i.e., CO_2 fugacity). For radium, the value is specified as a constant that depends on the range in which the pH value falls. For technetium, carbon, iodine, cesium, strontium, selenium, and chlorine, no concentration limit is applied; this abstraction, therefore, does not affect their release rates from the engineered barrier system." (SER Volume 3, pages 7-23 and 24)

"The NRC staff finds acceptable the applicant's overall approach to impose concentration limits because it is consistent with standard thermodynamic geochemical principles and uses consistent and appropriate assumptions. The use of pure-phase solubilities to constrain radioelement concentrations at the source is an accepted approach in performance assessments for radioactive waste disposal (e.g., Nuclear Energy Agency, 1997aa; Leslie, et al., 2007aa). When faced with uncertainty regarding the appropriate solid phase to model a solubility limit, the applicant appropriately chose the solid phase that would result in higher dissolved concentrations (e.g., hydrated PuO_2 instead of anhydrous PuO_2)." (SER Volume 3, page 7-24)

"The NRC staff finds that the applicant used appropriate tools to model concentration limits, including the important geochemical parameter inputs (e.g., pH and fCO_2) affecting the solubility model outputs. Concentration limits in the waste form domain are functions of chemical parameters developed by the in-package chemistry model (SER Section 2.2.1.3.4.3.1). Chemical conditions for concentration limits in the corrosion products domain were properly modeled using the surface complexation model, and the results of that model were supported by comparison with the applicant's independent modeling efforts in DOE (2009da, Enclosure 2). An abundant secondary phase, such as steel corrosion products, can have an important influence on pH buffering in an environment with such high solid-to-water ratios. In DOE (2009db, Enclosure 2), the applicant provided information that (i) supported its selection of the uncertainty range for the stainless corrosion rate on the basis of laboratory data and (ii) showed that plutonium isotope release rates, which are sensitive to the pH-dependent plutonium solubility limit, are insensitive to the stainless steel corrosion rate. The NRC staff evaluated the applicant's analysis by reviewing corrosion rate data in the literature (Beavers and Durr, 1990aa; BSC, 2004ae; Glass, et al., 1984aa; McCright, et al., 1987aa). In addition, the applicant showed in DOE (2009db, Enclosure 3) that conservatisms in (i) the treatment of the timing of radionuclide release after waste package breach, (ii) assumptions regarding flow within the waste package, and (iii) the lower pH range meant that the applicant did not overestimate the effectiveness of stainless steel corrosion products in controlling pH. The NRC staff finds that this conclusion is appropriate, in that the abstraction would not result in underestimation of radionuclide release rates, and is consistent with the NRC staff's understanding of the abstraction." (SER Volume 3, pages 7-24 and 7-25)

Colloids

"Colloids are 1- to 2-μm- [4 to 8 × 10^{-5}-in]-sized particles, have the potential to facilitate transport of highly sorbing, low-solubility radionuclides, and may allow radionuclide concentrations in water above their solubility limit. In the TSPA code, colloids in the engineered barrier system are formed by degradation of waste package internals and waste forms and also exist as groundwater colloids in seepage water entering breached waste packages. The applicant used the engineered barrier system colloid abstraction in the TSPA code to determine the stability and mass concentrations of reversible and irreversible colloid suspensions in the waste form, corrosion products, and invert domains of the engineered barrier system (SAR Section 2.3.7.11)." (SER Volume 3, pages 7-30 and 7-31)

"The NRC staff reviewed the TSPA model results for the disruptive modeling cases (those highly significant to risk) to evaluate processes and features that could limit availability and transport of colloid suspensions in the waste form, corrosion products, and invert domains of the engineered barrier system, as detailed in DOE (2009ay, Enclosure 3). The NRC staff finds acceptable the applicant's identification, description, and quantification of the distinct processes and features that control stable colloid concentrations in the engineered barrier system (e.g., considering the barrier capability of engineered barrier system components or the impact of seep and no-seep environments) for both the igneous intrusion and the seismic ground motion modeling cases. In addition to the simpler, long-term constant geochemical conditions considered in the igneous modeling case, the seismic ground motion modeling case demonstrated the effects of temporal variability in the stability and mass concentrations of colloids under both seep and no-seep conditions, as a function of patch failure developments on waste packages and resulting changes in ionic strength, pH, and relative humidity in the engineered barrier system.

The NRC staff conducted independent, simplified, confirmatory calculations on the effectiveness of iron oxide colloids in facilitating Pu-242 releases in the igneous intrusion modeling case. The igneous intrusion modeling case was chosen for independent calculations because (i) chemical conditions and colloid stability remain unchanged throughout the entire simulation after a relatively short cooling period (less than 1,000 years), (ii) this modeling case dominates the long-term total mean annual dose, and (iii) that dose is dominated by Pu-242 after 200,000 years. Using information provided by the applicant and used for Pu-242 in the TSPA model, the NRC staff's confirmatory calculations concluded that the ratio of (i) the plutonium attachment rate to iron oxide colloids to (ii) the plutonium attachment rate to stationary corrosion products is 2 × 10^{-7} (Pickett, 2010aa). These calculations showed that plutonium attachment to stationary corrosion products is much faster than attachment to iron oxide colloids.

As discussed in SER Section 2.2.1.3.4.3.5, the applicant showed in DOE (2009ay, Enclosure 8) that reversible, kinetic plutonium sorption onto stationary corrosion products can be approximated as an equilibrium process. Therefore, the rate of plutonium desorption from the stationary corrosion products is approximately equal to the rate of sorption. On the basis of this observation and the NRC staff's confirmatory calculation summarized in the previous paragraph, the NRC staff concludes that the rate of irreversible plutonium sorption to iron oxide colloids is many orders of magnitude slower than the rate of plutonium desorption from stationary corrosion products to solution. Therefore, any transfer of dissolved plutonium to iron oxide colloids would be compensated by desorption from the stationary corrosion products—which contain the majority of plutonium mass in the corrosion products domain—to maintain the quasi-equilibrium relationship. On the basis of its calculation, the NRC staff concludes that irreversible sorption of plutonium to iron oxide colloids cannot substantially deplete dissolved plutonium. Sorption to stationary corrosion products is more important to plutonium release from the corrosion products domain than are iron oxide colloids. This result is consistent with the applicant's conclusion that iron oxide colloids are not significant for Pu-242 releases from the engineered barrier system, as shown for representative realizations in DOE (2009dc, Figures 1.1-24 and 1.1-26) and DOE (2009da, Enclosure 1, Figures 5 and 7).” (SER Volume 3, pages 7-34 and 7-35)

EBS Transport

“The applicant’s abstraction for radionuclide transport in the engineered barrier system is highly significant to risk because large masses of plutonium and other dose-significant actinides are retained in the engineered barrier system in the applicant’s TSPA calculations. For example, in DOE (2009dc), the applicant provided results for a representative realization of the igneous intrusion modeling case showing that approximately 8,000 kg [17,600 lb] of Pu-242 is permanently immobilized in the engineered barrier system for one percolation subregion. In the same realization and subregion, approximately 30,000 kg [66,000 lb] of Np-237 is retained on the waste package corrosion products at 100,000 years; Np-237 is released from the engineered barrier system slowly enough that more than 1,000 kg [2,200 lb] remained at 1 million years.” (SER Volume 3, page 7-40)

“The applicant provided information showing that engineered barrier system releases of low-solubility, sorbing radionuclides (e.g., plutonium and neptunium) are mainly controlled by processes within the corrosion products domain because waste form dissolution and invert transport processes are fast, relative to transport within the corrosion products domain. In the TSPA analyses, the important dose contributions from plutonium and neptunium isotopes result from the igneous intrusion modeling case (SAR Section 2.4.2.2.1.1.3), in which all waste packages fail and releases of these radionuclides are controlled by advection modified by sorption and precipitation of radionuclide-bearing minerals.” (SER Volume 7-48)

“The NRC staff finds that, in modeling the transport of radionuclides in the engineered barrier system, the applicant adequately described the system and models used; applied appropriate conceptual models; and considered alternative conceptual models. The applicant used appropriate mathematical models to represent transport in the engineered barrier system. Transfer of information between the radionuclide transport abstraction and other TSPA code abstractions was consistently and appropriately implemented. Relevant design information for the waste package was appropriately incorporated. The applicant used appropriate data to establish model parameters and to represent uncertainty. Intermediate results of the abstraction were appropriately compared to independent information.

TSPA release rates for radionuclide transport in the engineered barrier system vary significantly by radionuclide and modeling case. The engineered barrier system does not significantly delay transport of soluble, nonsorbing radionuclides, such as Tc-99 and I-129, and the waste package failure rates control the engineered barrier system release rates for those radionuclides. Transport of low-solubility, sorbing radionuclides, such as Np-237 and Pu-242, is significantly slower and is generally controlled by sorption onto stationary corrosion products and precipitation of radionuclide-bearing minerals in the corrosion products domain. Colloid-assisted transport is not significant compared with transport of dissolved radionuclides because of limited colloid concentrations in the engineered barrier system. The NRC staff finds that the TSPA code results for the engineered barrier system release rates are consistent with the NRC staff’s simplified confirmatory calculations, confirming the appropriateness of the TSPA results.” (SER Volume 3, page 7-50)

6.3.5 Radionuclide Transport

DOE included transport processes in the unsaturated and saturated zones in estimating the releases from the repository to the accessible environment where potential exposures could occur. Consideration of transport processes included sorption of radionuclides on mineral surfaces as well as sorption onto colloids. Evaluation of unsaturated and saturated flow below the repository is also summarized briefly in this section.

Unsaturated Zone

The NRC review of the DOE abstraction of ambient flow in the unsaturated zone below the repository, which provides the flow framework for the unsaturated zone radionuclide transport abstraction, was summarized as follows:

> "On the basis of evaluations of the northern and southern portions of the proposed repository footprint, the NRC staff concludes that the range of flow fields generated from DOE's site-scale flow model adequately represents model and data uncertainty for performance assessment calculations. The NRC staff reaches this conclusion because (i) the resulting flow fields are unlikely to overestimate radionuclide travel times from the proposed repository to the water table and (ii) different parameter value sets either would minimally affect travel times or would increase travel times. Because the flow fields are directly used as input in the transport model abstraction, the NRC staff concludes integration between the unsaturated flow and transport abstractions is acceptable." (SER Volume 3 page 9-57)

The following SER excerpts touch on important aspects of the NRC review of unsaturated zone radionuclide transport.

> "In its Safety Analysis Report (SAR) Section 2.3.8 (DOE, 2008ab), DOE (i) described the features, events, and processes (FEPs) that DOE included to model the transport of radionuclides in groundwater in the unsaturated zone below the repository and (ii) provided the technical basis for DOE's implementation (or abstraction) of the unsaturated zone transport model in the Total System Performance Assessment (TSPA) model. The NRC staff's evaluation focuses on the following processes, detailed in subsequent sections, that DOE included in its SAR Section 2.3.8 as important for radionuclide transport in the unsaturated zone: (i) advection, because most of the radionuclide mass is carried through the unsaturated zone by water flowing downwards to the water table; (ii) sorption, because sorption in porous media in the southern half of the repository area has the largest overall effect on slowing radionuclide transport in the unsaturated zone; (iii) matrix diffusion in fractured rock, because matrix diffusion coupled with sorption slows radionuclide transport in the northern half of the repository area; (iv) colloid-associated transport, because radionuclides attached to colloids may travel relatively unimpeded through the unsaturated zone; and (v) radioactive decay and ingrowth, because these processes affect the quantities of radionuclides released from the unsaturated zone over time. The NRC staff's review of DOE's technical basis for excluding other FEPs is addressed in the SER Section 2.2.1.2.1 (Scenario Analysis)." (SER Volume 3, page 10-1)

> "In DOE's unsaturated zone transport abstraction, the migration of radionuclides through the unsaturated zone is influenced by the transport-affecting processes of (i) advection and dispersion, (ii) sorption, (iii) matrix diffusion, (iv) colloid-associated radionuclide transport, and (v) radioactive decay and ingrowth (SAR Section 2.3.8.1). Advection, dispersion, matrix diffusion, and colloidal transport are transport mechanisms that move radionuclides from one location to another. In contrast, sorption may delay the transport of a radionuclide by attachment to stationary surfaces such as the rock matrix." (SER Volume 3, page 10-7)

> "Flow path differences between the northern and southern portions of the repository influence the travel times of non-sorbing and sorbing radionuclides. DOE provided model results (SAR Figures 2.3.8-36 and 2.3.8-49) that showed three predominant types of transport pathways. These are (i) fast transport for fracture releases occurs in the northern half of the repository, with mean travel times of years to centuries; (ii) moderately slow transport pathways for both matrix and fracture releases go through the southern half of the repository, with mean travel times of centuries to millennia; and (iii) slow transport through the matrix for radionuclides released into the matrix of the TSw tuff with mean travel times of millennia, with a small percentage transferring to the fracture system and reaching the water table more rapidly. The DOE ambient site-scale unsaturated zone model includes perching below the repository horizon in the northern half of the repository. In the DOE implementation, perching diverts fracture waters into faults and thereby creates a large difference in travel times for the northern and southern halves of the repository."
> (SER Volume 3, page 9-56)

“On the basis of its review and the NRC staff’s knowledge and experience (e.g., Bertetti, et al., 2011aa; Turner, et al., 2002aa), the NRC staff concludes that DOE adequately incorporated sorption modeling in performance assessment calculations for the following reasons:

- DOE based its sorption modeling on an empirical K_d modeling approach that is well established (e.g., Freeze and Cherry, 1979aa; Till and Meyer, 1983aa) and has been broadly used to describe radionuclide transport (e.g., Sheppard and Thibault, 1990aa; Chapman and McKinley, 1987aa).

- DOE defined and documented the limitations of the K_d approach and used stochastically sampled K_d probability distributions and simplifying assumptions about the effectiveness of sorption to address model and data uncertainty.

- DOE considered the range of expected site geochemical and physical conditions in developing the K_d probability distributions, and addressed uncertainty by using either low K_d values or bounding assumptions that reduce the credit given to radionuclide sorption in the TSPA model.

- DOE adequately described how it obtained, used, and interpreted experimental data with site-specific materials, alternative computer models, field tests, and natural analogs to provide a technical basis to support the TSPA model abstraction of radionuclide sorption.

- DOE considered alternative sorption modeling approaches and used them to support the technical basis for the K_d distributions.

- DOE adequately described the method used to assess the sensitivity of radioelement sorption behavior to variability in geochemical and physical conditions, and DOE acceptably used that method to correlate sorption characteristics among the radioelements, ensuring consistency among the sorption parameters for each TSPA model realization.

- DOE identified potential sources of uncertainty on the basis of site- and radionuclide-specific data and propagated the uncertainty through the unsaturated zone transport model abstraction by using lower, limited ranges of K_d values. With respect to the TSPA model abstraction, this underprediction means that DOE takes less credit for sorption in the unsaturated zone than experimental results would indicate.

- DOE used observations from natural analogs to support model abstraction and uncertainty by constraining sorption processes in unsaturated fractured rock.

- DOE reduced the significance of model uncertainty of radionuclide transport in unsaturated zone fractures by taking no performance assessment credit at all for sorption in the fractures.”
  (SER Volume 3, pages 10-12 and 10-13)

“The NRC staff reviewed DOE’s technical basis for the colloid-associated transport model in the context of the NRC staff’s independent understanding of colloid-associated transport modeling, colloid stability, and colloid transport properties in natural and engineered systems. As DOE noted in SAR Section 2.3.8.3, colloid transport mechanisms in unsaturated, fractured rocks are not well characterized by field studies. Accordingly, the NRC staff’s review of DOE’s technical basis for colloid-associated transport of radionuclides in the unsaturated zone focuses on how DOE addressed data and model uncertainty in developing parameter values and modeling colloid-associated transport processes. The NRC staff evaluated information DOE provided in SAR Section 2.3.8 and references therein, particularly SNL [2008ag, Section 7.7.1(a)], SNL (2007bi, Section 6.3.1), and SNL (2008an). The NRC staff reviewed the DOE integration of colloid-associated transport between the EBS, the unsaturated zone, and the saturated zone by examining SAR Section 2.3.7 (Waste Form Degradation and Mobilization and Engineered Barrier

System Flow and Transport), SAR Section 2.3.9 (Saturated Zone Flow and Transport), and supporting references. The NRC staff also considered additional information that DOE provided to clarify details of the colloid-associated transport model in DOE (2009am, Enclosures 9 through 14). Based on its review, the NRC staff concludes that the data and methods DOE used to estimate unsaturated zone transport parameters for reversible colloids are acceptable for the following reasons:

- DOE compensated adequately for a scarcity of unsaturated zone colloid transport data by using data from saturated zone groundwater analyses and Yucca Mountain saturated zone colloid transport tests to estimate unsaturated zone colloid properties.

- By incorporating the available site-specific data to set initial and boundary conditions for colloid properties, DOE's colloid-associated transport model adequately accounted for system variability and included sufficient data to describe colloids in the natural system.

- DOE addressed data uncertainty adequately by (i) sampling large ranges for colloid-associated parameter values to account for data uncertainty about natural colloid properties and (ii) sampling the ranges of parameter values separately for the unsaturated zone and saturated zone transport abstractions to account for data uncertainty and spatial heterogeneity in the natural system."
(SER Volume 3, page 10-19)

Saturated Zone

The NRC review of the DOE abstraction of flow in the saturated zone, which provides the flow framework for the saturated zone radionuclide transport abstraction. Once in the saturated zone, groundwater flows through a volcanic rock aquifer in the northern portions of the general flow system, transitioning into an alluvial aquifer system in the southern portions of the Yucca Mountain region. The location at which groundwater flows from fractured volcanic rocks to alluvium is significant because of the differences in the hydrologic properties between these two rock units. DOE model results indicated that sorption within the alluvium effectively delays the transport of moderately and strongly sorbing radionuclides for thousands of years or longer (SAR Sections 2.3.9 and 2.1.2.3.6). The NRC evaluated DOE's approach to treat uncertainty associated with the tuff/alluvium contact and found: "(i) DOE reasonably bounded the extents of the alluvium uncertainty zone and (ii) the uniform distributions defined for parameters representing western and northern extents of alluvium reasonably propagate uncertainties associated with the actual geometry of the volcanic and alluvium contact" (SER Volume 3, page 11-14) Sorption processes in the saturated zone, including colloidal transport, are discussed below.

"Radionuclide transport in the saturated zone, as described in SAR Section 2.3.9, includes the features, events, and processes (FEPs) that affect the movement of radionuclides from where they enter the saturated zone below the repository to the accessible environment boundary approximately 18 km [11.18 mi] south of the repository and their implementation (or abstraction) in the TSPA." (SER Volume 3, page 12-1)

"Sorption, as stated in SER Section 2.2.1.3.7.3.2.2, is a general term for chemical and physical processes that transfer a fraction of a dissolved species to the surface of a solid phase. Depending on specific properties of the dissolved species, the solid phase, and the liquid phase, the extent of sorption varies: some radionuclides will sorb strongly to the solid, some will sorb weakly onto the solid, and some will not sorb at all. As modeled by DOE for the transport of radionuclides through the saturated zone, sorption onto the fractured volcanic tuff matrix or onto alluvium results in retardation, or slowing, of radionuclides relative to rates of water flow through the saturated zone. In contrast, radionuclide sorption onto mobile colloids may enhance the transport rate of radionuclides relative to their sorption onto a stationary solid.

DOE identified sorption as an important process contributing to the barrier capability of the saturated zone (SAR Section 2.3.9). In particular, DOE model results indicate that sorption within the alluvium effectively delays the transport of moderately and strongly sorbing radionuclides for thousands of years or longer (SAR Sections 2.3.9 and 2.1.2.3.6). DOE estimated that sorption of dissolved thorium, americium, and protactinium is so effective in the saturated zone that, upon entering the saturated zone, these radionuclides cannot traverse it to reach the accessible environment within the regulatory period of 1 million years. For these radionuclides to be present at the accessible environment boundary within the million-year timeframe, DOE determined that they must either be transported through the saturated zone as colloids or be ingrown as the decay products of mobile parents." (SER Volume 3, page 12-12)

"DOE noted that the primary controls on sorption are (i) the characteristics of the mineral surfaces onto which sorption occurs, (ii) the chemistry of groundwater in the saturated zone, and (iii) the sorption characteristics of each element (SAR Section 2.3.9.3.2.2). DOE assumed sorption of dissolved radionuclides would occur only in the matrix of the volcanic tuff or in the alluvium. Citing uncertainties about the nature of the fracture coatings, DOE excluded sorption onto fracture surfaces in the volcanic rock (SAR Section 2.3.9.3.2.2; SNL, 2007ba). However, DOE did assume that solutes transported through designated fault or fault-related fracture zones could undergo sorption depending on the characteristics of the zone (BSC, 2005ak). In fault-related fracture zones, a small portion of the rock matrix within the fracture zone was conceptualized as allowing rapid diffusion, and a retardation factor was calculated accordingly (BSC, 2005ak). DOE also assumed that mobile colloids could be retarded within fractures of the volcanic tuff (SAR Section 2.3.9.3.3). DOE cited laboratory and field-scale transport experiments to support its conceptual model of colloid retardation in fractures (BSC, 2005ak; SNL, 2007aw)." (SER Volume 3, page 12-12)

"The NRC staff reviewed the geochemical controls on radionuclide sorption and the experimental data DOE used to develop the TSPA K_d distributions (Bertetti, et al., 2011aa). The NRC staff finds that the major ion chemistry (e.g., calcium, sodium, bicarbonate) of the waters used in DOE sorption experiments is comparable to that of saturated zone waters, as described in SNL (2007ba, Appendix A). The UE-25, J-13, and UE-25 p#1 water chemistries bound the ranges reported for saturated zone water chemistries for major ions such as sodium, calcium, and bicarbonate, and other parameters such as pH and redox state. Based on the NRC staff's knowledge and experience, these chemical characteristics are likely to be the most important for radionuclide sorption (e.g., Turner and Pabalan, 1999aa). The NRC staff further concludes that the chemistries of alluvial aquifer waters used in alluvium sorption experiments are representative of conditions in the alluvium (McMurry and Bertetti, 2005aa).

The NRC staff concludes that the experimental approaches used by DOE to develop K_d values are adequate. In selecting experimental data to inform the TSPA K_d distributions, DOE appropriately excluded data from experiments where the final radionuclide concentration indicated that the solubility limit of the radionuclide may have been exceeded, as described in SAR Section 2.3.9.3.2, SNL (2007ba, Appendix A), and SNL (2007ah), and DOE did not include a particular set of experiments that may have been conducted at initial concentrations that were above the solubility limits for some radionuclides [SNL (2007ba), Appendix A]." (SER Volume 3, page 12-14)

"DOE modeled colloidal transport in the saturated zone consistent with modeling used elsewhere in the TSPA, with two types of radionuclide attachment: reversible and irreversible (BSC, 2005ak). Colloids with irreversibly attached radionuclides were modeled as separate transported entities, with a retardation factor applied specifically to the fractured volcanic tuff and alluvial aquifers to simulate the effects of nonpermanent filtration. DOE assumed that the size of irreversible colloids could exceed that of the pores of the volcanic matrix. Consequently, DOE did not incorporate matrix diffusion of irreversible colloids in the saturated zone transport abstraction. Plutonium and americium were modeled as associated with both irreversible colloids and reversible colloids and as dissolved species in the saturated zone transport model, consistent with DOE's unsaturated zone transport model. Reversible colloidal transport was modeled using the K_c factor, which represented equilibrium sorption of aqueous radionuclides onto natural system colloids. Radionuclides associated with reversible colloid transport comprised 4 of the 12 radionuclide groups modeled in the saturated zone flow and transport abstraction. These four groups included (i) plutonium, (ii) cesium, (iii) tin, and (iv) americium, protactinium, and thorium. Application of the K_c factor and inclusion of reversible sorption to colloids lowered the effective diffusion coefficient and the sorption coefficient, K_d, for the radionuclides, enhancing advective transport." (SER Volume 3, page 12-20)

"On the basis of the information DOE provided in SAR Section 2.3.9 and supporting references, the NRC staff concludes that DOE's representation of colloid-associated transport in the saturated zone is acceptable for performance assessment calculations for the following reasons:

- DOE developed an adequate conceptual and mathematical basis for colloid-associated transport processes in the saturated zone (e.g., retardation of colloids by attachment processes in fractured volcanic tuff and alluvium, reversible sorption of radionuclides onto colloids, colloid exclusion processes, and unretarded colloidal transport) that is consistent with existing models for contaminant transport in fractured rocks and porous media in the literature (e.g., Sudicky and Frind, 1982aa).

- DOE provided model results that are consistent with cross-hole field tests using microspheres showing decreased retardation of colloid-associated radionuclides relative to dissolved constituents. The modeling results and field-test results are consistent with the K_c factor approach used to represent colloid-associated transport.

- DOE selected a set of radioelements to model colloidal-facilitated transport that are the most strongly sorbed, and the saturated zone approach is consistent with that used in DOE's unsaturated zone model. The radioelements that are the most strongly sorbed to the colloids are those that contribute the most to dose.

- DOE's treatment of colloid-associated transport is consistent with DOE's model for partitioning of the radioelements among the three transport entities (dissolved species, reversibly associated with colloid, and irreversibly associated with colloid), which is evaluated in SER Section 2.2.1.3.4. The NRC staff considers the inclusion of similar colloid-associated modeling approaches and assumptions used for parent and daughter radionuclide attachment in both saturated zone transport models to be adequate and consistent.

- DOE's assumptions for colloid concentrations and stability in the saturated zone are consistent with groundwater analyses observations for the Yucca Mountain region. Although naturally occurring colloids in Yucca Mountain groundwaters consist of montmorillonite, zeolite, and silica, the use of montmorillonite alone is adequate, as the specific mineral is less significant than the sorption coefficients assigned to it. DOE broadly addressed data uncertainty for sorption onto reversible colloids by selecting a reasonable range of montmorillonite sorption coefficients, which captures the sorption behavior of other potential colloid minerals.

- DOE adequately accounted for radioactive decay and ingrowth processes for radionuclides in the form of dissolved species, reversible colloid species, and irreversible colloid species included in the saturated zone transport abstraction. DOE's treatment of decay chain daughter nuclides for irreversible colloids is adequate because it is consistent with DOE's model assumptions about which radionuclides are associated with reversible and irreversible colloids.

- DOE's modeling approach adequately compensated for the high uncertainty in empirical observations for saturated zone colloidal transport in field studies or natural analogs by using reasonable probability distributions for most colloid-related parameters.

With respect to DOE's representation of radionuclide transport by reversible colloids, the NRC staff concludes that DOE provided an adequate technical basis by accounting for system variability in developing parameter values, where feasible, from site-specific data from saturated zone field tests in the Yucca Mountain area and sampling colloid-associated parameter values from large uncertainty distributions. The NRC staff finds that DOE adequately addressed model uncertainty because the results are consistent with the NRC staff's understanding of colloid-associated transport processes and the uncertainties involved in characterizing colloidal transport processes in natural systems.

With respect to DOE's representation of radionuclide transport by irreversible colloids, the NRC staff concludes that DOE's model is adequate because it includes processes that have been demonstrated to be present in field tests and lab experiments. DOE's approach used reasonable distributions of parameter values, simple model abstractions supported by field and lab tests, and analyses of natural analogs and underground nuclear tests. For example, the only radioelements irreversibly associated with colloids in DOE's model are plutonium and americium; this assumption is integrated with DOE's near-field model assumptions stating that after the failure of waste containers due to general corrosion in TSPA simulations, up to 30 percent of the Pu-242 flux transported to the accessible environment is by irreversible colloids (e.g., SAR Section 2.4.2.2.3.2.2 and Figure 2.4-108). The NRC staff concludes that DOE's election to not consider permanent filtration of irreversible colloids is acceptable because it allows for larger releases of colloid-associated radionuclides." (SER Volume 3, pages 12-21 and 12-22)

6.3.6 Igneous Activity

"DOE examined the consequences of igneous disruption of the repository (Igneous Scenario Class) using the results of TSPA calculations through the two linked modeling cases: igneous intrusion and volcanic eruption (intrusion always precedes eruption). DOE's igneous intrusion modeling case provides TSPA parameter values for the number of waste packages failed (mass of waste) during an intrusive event, the temperature in the invaded drifts in the period after intrusion, and chemical changes to groundwater that may react with the basalt filling the drifts. The igneous disruption of waste packages abstraction integrates with other TSPA model components, such as the unsaturated zone radionuclide transport abstraction, and provides information about the flux of radionuclides released from the waste form into water entering the unsaturated zone after an intrusive event (SER Section 2.2.1.3.7). Exposure to radionuclides in groundwater extracted by pumping is one of the principal pathways for radiological exposure to the reasonably maximally exposed individual (RMEI). In the DOE volcanic eruption modeling case, a key parameter affecting the overall radiological dose calculation is the number of directly affected waste packages and thus the amount of waste entrained in a volcanic eruption." (SER Volume 3, page 13-1)

"DOE explained that the volcanic (extrusive) part of the igneous scenario is an extension of the intrusive part (SAR Section 2.3.11.1) and concluded that every intrusive event that might intersect the repository is likely to have a conduit develop somewhere along one of the dikes, as described in SAR Section 2.3.11.2.1.2 and SNL Table 7-1 (2007ae). The conduit (or conduits) may, however, form outside the repository footprint or may not intersect a drift, and in that case, no waste material would be entrained into the magma that rises to the surface in an eruption. In effect, this would be equivalent to the intrusive-only case. In addition, DOE determined that conduits that might feed surface volcanoes may only develop along specific parts of dikes (SAR Section 2.3.11.4.2.1) and thus concluded that the probability of a volcanic event occurring at the repository is expected to be lower than the probability of an intrusive event. DOE also concluded that if an eruption that entrained waste material and transported it into the surface environment did occur at the repository, the potential doses to the RMEI location from radionuclides released through the intrusive and extrusive pathways would be additive. Further details of conduit development are evaluated in the NRC staff's review of the volcanic eruption modeling case (SER Section 2.2.1.3.10.3.3)." (SER Volume 3, page 13-5)

"NRC staff has assessed the risks caused by an igneous event at the proposed repository on the basis of the applicant's information and the NRC's independent analysis. As stated in the Introduction (SER Section 2.2.1.3.10.1) of this section, while the probability of an igneous event is low, the consequences could be high. The igneous intrusion modeling case would constitute most of the calculated dose to the RMEI for the first 1,000 years following permanent closure of the repository, as shown in SAR Figure 2.4-18(a), and is approximately half the DOE calculated dose for the seismic ground motion modeling case in the ensuing 9,000 years.

In SAR Section 2.4.2.2.1.2.3, DOE provided its estimate of probability-weighted consequences of igneous activity (intrusive and extrusive) using the probability distribution from its expert elicitation for a Probabilistic Volcanic Hazard Assessment (PVHA). DOE identified that the probability-weighted igneous mean intrusive dose is estimated to be less than 0.001 mSv/yr [0.1 mrem/yr] for the 10,000-year period and the median dose less than 0.005 mSv/yr [0.5 mrem/yr] for the post-10,000-year time period (SAR Section 2.4.2.2.1.2.3.1). DOE estimates for the probability-weighted igneous extrusive (volcanic eruptive) mean dose alone are on the order of 10^{-6} mSv/yr [0.0001 mrem/yr] for the 10,000-year period and the median dose is less than 6×10^{-7} mSv/yr [6×10^{-5} mrem/yr] for the post-10,000-year time period (SAR Section 2.4.2.2.1.2.3.2). The NRC staff notes that the difference in magnitude for the dose consequences between the two igneous scenarios (intrusive and extrusive) predominantly results from the different number of waste package failures estimated to occur for each scenario, which causes the dose from the extrusive case to be several orders of magnitude below the intrusive case (SAR Section 2.2.1.4.1; evaluated in SER Section 2.2.1.3.10.3.3." (SER Volume 3, page 13-6)

Intrusive Igneous Case

"In the intrusive igneous case, DOE assumes that if a single rising dike intersects any part of the repository footprint where drifts containing waste packages are located, then all drifts in the repository are rapidly filled with magma. DOE developed this approach to account for the uncertainties in determining the physical characteristics of dikes at repository depths, and for uncertainties in magma flow processes in drifts intersected by dikes (SAR Section 2.3.11.3.1). For the ascending magma entering the drifts, DOE recognized that there are two possibilities for flow behavior, considering how rapidly and violently magma could enter a drift. The less rapid possibility is effusive and has a lava-like flow, while the other is more explosive, resulting in a fragmental, or pyroclastic flow (SAR Section 2.3.11.2.1.2; SNL, 2007ag; Woods, et al., 2002aa; Dartevelle and Valentine, 2005aa, 2009aa). The NRC staff also conducted independent confirmatory analyses (Woods, et al., 2002aa; Lejeune, et al., 2009aa) verifying that potential magma flow into drifts could occur quickly enough so that only minor cooling of the magma would occur. On the basis of the results of these independent studies and its own evaluation, the NRC staff concludes that DOE has developed an acceptable technical basis to propose that all drifts will be filled with basaltic magma if an intrusive igneous event occurs at the repository site, as further discussed in this section. This approach involves the disruption of all waste packages stored in the proposed repository (SAR Section 2.3.11.3). The NRC staff further concludes that this does not underestimate risk and that there are no technical uncertainties in this conclusion that could reasonably increase the DOE risk estimates."
(SER Volume 3, page 13-8)

Extrusive Igneous Case

"DOE proposed that in some potential igneous intrusive events that intersect the repository footprint, a rising dike would reach the surface and develop a conduit at a location along the intrusion, and that magma would be extruded. If a conduit is located wholly or partially in a repository drift, waste from disrupted waste packages could be entrained by magma flow up the conduit and erupted from a volcano at the surface. Compared with the intrusion scenario, in which the contents of all waste packages in the repository are made available for hydrologic transport, DOE concluded that, for the volcanic scenario, only a limited amount of high-level waste could be entrained directly into a conduit or conduits (SAR Section 2.3.11.4), as explained next.

In the type of basaltic volcanic activity DOE predicted for the case of an eruption through the proposed repository, a dike reaches the surface and activity begins along a fissure (an elongated system of vents, which is the surface expression of the dike; see SAR Sections 2.3.11.2.1 and 2.3.11.4.1.1 and SAR Figure 2.3.11.5). In DOE's model, magma flow to the surface in the dike usually localizes to a single, or a few, points over a period of hours to a few days, as observed at past basaltic eruptions and as previously discussed in SER Section 2.2.1.3.10.3.2. Such behavior was seen in analogous historic events [e.g., the

1943–1952 eruption of Paricutín in Mexico, the 1973 Heimaey eruption in Iceland, and the 1975 Tolbachik eruption in Kamchatka (Pioli, et al., 2008aa; Thorarinsson, et al., 1973aa Doubik and Hill, 1999aa)]. DOE studies of igneous products exposed in the rock record also inferred a similar progression for some prehistoric basaltic eruptions (e.g., SAR Section 2.3.11.4; SNL, 2007ae; Valentine, et al., 2006aa; Keating, et al., 2008aa). At this point in the modeled eruption, a conduit is considered to develop below the point of localization, with the main vent at the surface. This conduit and vent system feeds an explosive and lava-flow-forming Strombolian-style eruption. DOE adopted a violent Strombolian style for the entire model eruption considered on the basis of the characteristics of the young Lathrop Wells scoria cone near Yucca Mountain (see SER Section 2.2.1.2.2). DOE recognized that conduits grow (widen) downwards from the surface in the plane of the dike, as detailed in SAR Section 2.3.11.4.2.1.2 and SNL p. 6-46 (2007ae), and thus, in DOE's repository-disruption scenario, intersect a drift through the top of the drift.

DOE characterized subsurface volcanic conduits as flaring inward down from the top of the surface vent, such that conduit diameters at repository depths will be smaller than those observed near the surface. DOE characterized the size and shape of conduits using studies at exposed local analogous volcanoes (SAR Section 2.3.11.4 and Figure 2.3.11-6; SNL, 2007ae; Valentine, et al., 2006aa; Keating, et al., 2008aa) and theoretical considerations and model studies (e.g., Wilson and Head, 1981aa; Valentine, et al., 2007aa). In the performance assessment, DOE represents subvolcanic conduits as simple cylinders (SAR Section 2.3.11.4.1). DOE used the area of the conduit that intersects a drift to calculate the mass of waste that the conduit entrains. DOE concluded that entrained waste is mixed uniformly in the volume of magma that is subsequently erupted at the surface. From a risk perspective, the DOE performance assessment calculates that the expected annual dose from the igneous volcanic modeling case alone is approximately 0.1 percent of the dose calculated for the intrusive scenario (SNL, 2007ag). This difference between the volcanic and intrusive scenarios arises, in part, because DOE concluded that the volcanic scenario entrains and erupts approximately 0.1 percent of the amount of high-level waste that is disrupted during the intrusive case. The NRC staff finds this conclusion to be acceptable, as detailed next. Thus, the NRC staff's review of the subsurface processes associated with the volcanic case focuses on the DOE basis for concluding that a volcanic conduit, or conduits, would entrain a limited amount of waste."
(SER Volume 3, pages 13-13 and 14)

"The NRC staff reviewed DOE information regarding the likelihood for conduit development at repository drifts. From studies of the characteristics of basaltic volcanism at the Yucca Mountain region and elsewhere (Hill and Conner, 2000aa; Doubik and Hill, 1999aa) and DOE and independent confirmatory and external studies of conduit development in basaltic volcanism (BSC, 2003ab; Detournay, et al., 2003aa; Pioli, et al., 2008aa), the NRC staff concludes that DOE acceptably characterized the number and spacing of volcanic conduits. The NRC staff finds that the DOE conclusion that the processes leading to conduit development along a dike are reasonably represented as randomized along the widest dike segment. This is acceptable because there is no predictable pattern controlling conduit formation at other analogous basaltic volcanoes. The NRC staff reviewed the DOE methodology that developed the 28 percent factor for conduit development in the repository and the 30 percent factor for conduit intersection with a drift. The NRC staff concludes that DOE acceptably implemented randomized conduit development in developing these factors, and that even if the conduit development factor was significantly higher, the implied risk would change by only a small amount (e.g., using a factor of 100 percent would increase the amount of waste disrupted and ejected to ~0.3 percent of that disrupted in the intrusive case, versus the predicted value of 0.1 percent). Given the relatively small volume and rapid infilling time of the intersected drifts, the NRC staff concludes that the presence of repository drifts will not significantly affect the localization process for conduit development. Thus, the NRC staff concludes that DOE has acceptably evaluated the likelihood of conduit development at intersected drifts." (SER Volume 3, page 13-15)

6.3.7 Estimating Overall Performance

"DOE has identified three distinct event scenario classes (also referred to as event classes or scenario classes) that are included in its TSPA model to demonstrate compliance with the individual protection standard: (i) early failures, (ii) seismic events, and (iii) igneous events. DOE has used two modeling cases within each scenario class to represent specific aspects of the scenario. The early failure scenario class is composed of an early waste package failure modeling case and an early drip shield failure modeling case. The seismic scenario class is composed of a seismic ground motion modeling case and a seismic fault displacement modeling case. The igneous scenario class is composed of an igneous intrusion modeling case and a volcanic eruption modeling case." (SER Volume 3, page 17-3)

"The DOE TSPA assessment incorporates the following three distinct event scenario classes: (i) the igneous activity scenario class, which has a very low annual probability [on the order of a 1 in 100 million chance of occurring per year, as outlined in CRWMS M&O (1996aa)]; (ii) the seismic scenario class, which typically results in numerous events occurring over 1 million years (according to SAR Section 2.4.2.1.6, p. 2.4-50, seismic events are expected to occur frequently; however, it is important to evaluate the timing and magnitude of seismic events); and (iii) the early failure scenario class, for which there is a low probability of occurrence for an individual waste package (SAR Section 2.4.2.1.6, p. 2.4-49). These three event scenario classes include the occurrence of nominal processes, whereas, the nominal scenario class represents repository behavior in which no events occur (i.e., no seismic events, no igneous events, and no early failure events; see SAR Section 2.4.2.1.3, pp. 2.4-30-31). The applicant has described how its approach to combine the scenarios to derive aggregated annual dose estimates is appropriate in that it tends to slightly overestimate dose by double counting waste packages potentially affected by different failure modes from the different scenarios (e.g., waste packages failed by a seismic event and an igneous event would be double counted; see SAR Section 2.4.2.1.7)." (SER Volume 3, page 17-4)

Figure 6-1 presents the relative contributions of DOE's modeling cases to the total mean annual dose. The NRC staff performed a confirmatory calculation to assists the review of the DOE's TSPA models and calculations.

"Overall, the annual doses from the confirmatory calculation are in general agreement with the TSPA results (e.g., a majority of either single value comparisons are within a factor of two or single values for the TSPA results fall between the upper and lower values when the confirmatory calculation provides both an upper and lower value). The igneous intrusive modeling case, which is already somewhat simplified in the TSPA model by assuming all waste packages fail when the event occurs, tends to exhibit the best fit between the confirmatory calculation and the TSPA results. The fit for Tc-99 also exhibits a better fit regardless of the modeling case because the representation of Tc-99 in the repository is less complex: high solubility and mobility for Tc-99 limits the factors affecting release and transport of Tc-99. Although the ground motion modeling case is a bit more complicated due to the variety and timing of waste package breaches (e.g., cracks, ruptures, and patches), the results of the NRC staff's confirmatory calculation are in general agreement with the TSPA results. There is no precise agreement between the NRC staff's confirmatory calculation and the results of the DOE TSPA results due to the simplifying assumptions made in the confirmatory calculation [see NRC and CNWRA (2014aa) for further details on assumptions for the NRC staff confirmatory calculation]. The NRC staff's confirmatory calculation was used to confirm the NRC staff's understanding of the key attributes of the repository performance in the DOE TSPA analyses and to confirm that those attributes are consistent with DOE dose results. The confirmatory calculation considered the effect on dose by (i) the number of waste packages and the extent of waste package damage; (ii) the drift seepage; (iii) solubility limits for individual radionuclides (including the effect of corrosion products); (iv) the inventory of specific radionuclides; (v) sorption in the geosphere; and (vi) the probability of disruptive events. Consistency between the confirmatory calculation and DOE TSPA results provides further confidence that DOE TSPA analysis is consistent with the model abstractions described in the license application and reviewed in SER Section 2.2.1.3." (SER Volume 3, page 17-26)

The NRC review also considered the effect of uncertainties in the TSPA results, for example:

> “The effect of uncertainties on the DOE average annual dose curve is limited primarily because a number of important aspects of repository performance are near maximum values. For example, after 10,000 years nearly all the waste packages are dripped on (seepage fraction of 69 percent for seismic ground motion modeling case and 100 percent for igneous intrusion; see Table 17-2) and, given the 1-million-year period, a variety of long-lived radionuclides can eventually make it to the RMEI location (see Tables 17-4 and 17-5). Releases from the waste package will be affected by the failure rate for the waste package, including the areal extent of the waste package breaches, solubility limits, and the effect of corrosion products. The confirmatory calculation considered the low and high values of the solubility limits to provide some insight on how uncertainty in release from the waste package might impact the annual dose. Use of the highest solubility limit, as expected, increases the annual dose in the NRC staff’s confirmatory calculation. The estimated peak dose in the NRC staff’s confirmatory calculation at 10,000 years is 0.0014 mSv/yr [0.14 mrem/yr] compared to the regulatory limit of 0.15 mSv/yr [15 mrem/yr] for the initial 10,000 years and an estimated peak dose of 0.025 mSv/yr [2.5 mrem/yr] at 800,000 years compared to the regulatory limit of 1.0 mSv/yr [100 mrem/yr] for the period after 10,000 years.” (SER Volume 3, page 17-29)

6.3.8 Conclusions

Volume 3 of the SER concludes:

> “The NRC staff has reviewed and evaluated the DOE’s Safety Analysis Report, Chapter 2: Repository Safety After Permanent Closure and the other information submitted in support of its license application and has found that DOE submitted applicable information required by 10 CFR 63.21. The NRC staff has also found with reasonable expectation, that (i) the proposed Yucca Mountain repository design meets the applicable performance objectives in Subpart E, including the requirement that the repository be composed of multiple barriers and (ii) based on performance assessment evaluations that are in compliance with applicable regulatory requirements, meets the 10 CFR Part 63, Subpart L limits for individual protection, human intrusion, and separate standards for protection of groundwater.” (SER Volume 3, page xxxi)

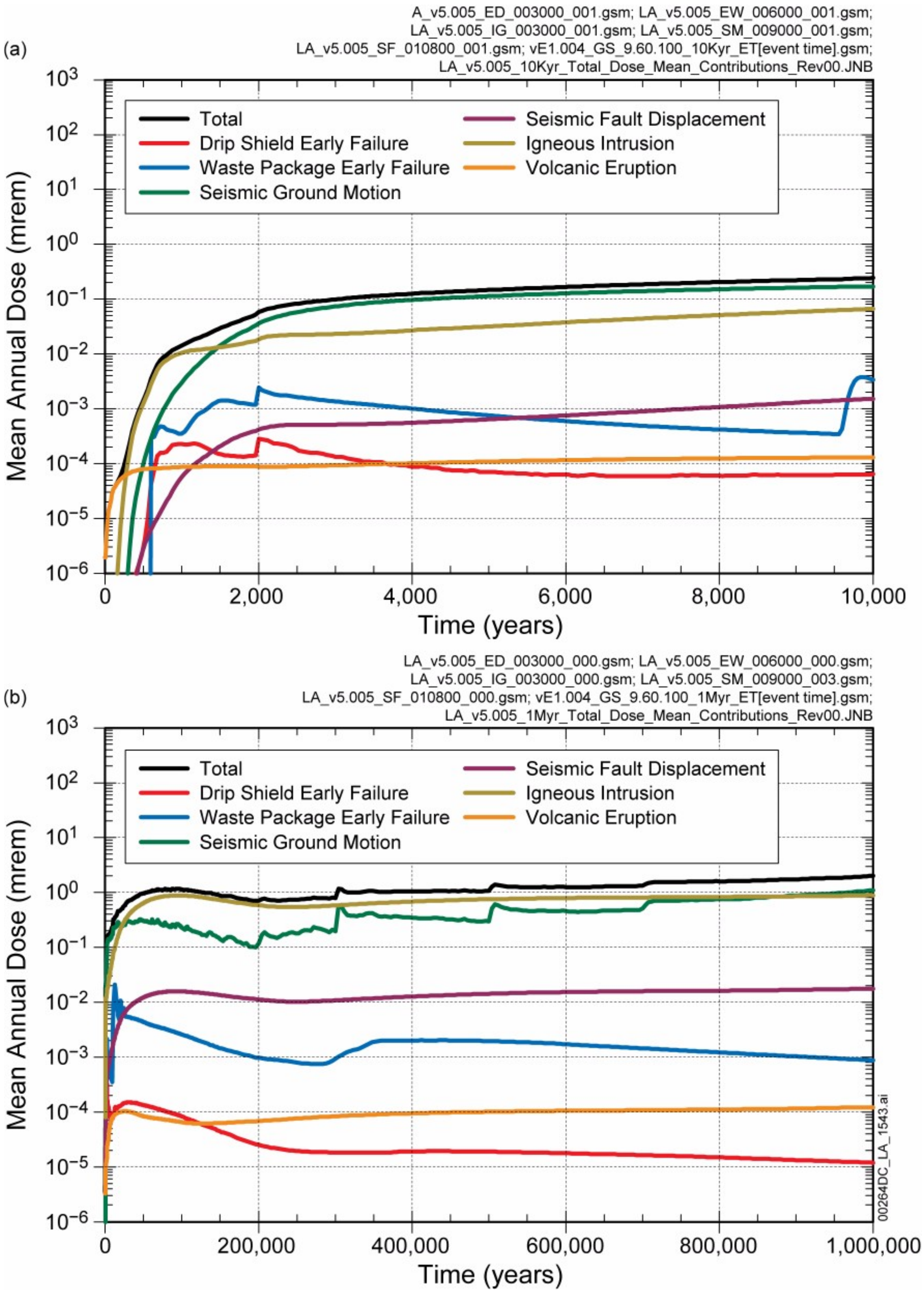

Figure 6-1. Relative Contributions of Modeling Cases to Total Mean Annual Dose for (a) 10,000 Years and (b) 1 Million Years after Repository Closure (SAR Section 2, Figure 2-4.18)

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6.3.9 References

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6.4 Administrative and Programmatic Requirements (NUREG-1949 – SER Volume 4)

The Administrative and Programmatic Requirements reviewed entailed: research and development to resolve safety questions; performance confirmation program; quality assurance program; records, reports, tests, and inspections; DOE's organizational structure; key safety positions; personnel qualifications and training requirements; plans for startup activities and testing; plans for conduct of normal activities, including maintenance, surveillance, and periodic testing; emergency planning; controls to restrict access and regulate land uses; and uses of the GROA for purposes other than disposal of radioactive wastes. The areas of the performance confirmation program, the quality assurance program, and controls to restrict access and regulate land uses are described below as these areas represent some of the more significant areas of the administrative and programmatic review. Although not discussed here in a separate subsection, the NRC review regarding the required research and development program to address safety questions led to a proposed condition for construction authorization (see Section 6.4.4).

6.4.1 Performance Confirmation Program

> "The Performance Confirmation Program is the set of tests, experiments, and analyses that are conducted, where practicable, to evaluate the adequacy of the assumptions, data, and analyses supporting DOE's application to construct and operate a high-level waste (HLW) repository at Yucca Mountain. The objective of performance confirmation is to monitor key geotechnical and design parameters, including interactions between natural and engineered systems and components throughout construction, operation, and through to closure, to identify significant changes from the conditions assumed and evaluated in the license application that would affect postclosure safety. Changes in parameters and conditions during construction, operation, and through to closure are identified by the Performance Confirmation Program by comparison with baseline and expected values. Baseline values are developed for the Performance Confirmation Program using assumptions, data, and analyses that DOE provided in its SAR to support the license application, and that the NRC evaluates in making its licensing decision.
>
> The performance confirmation program does not confirm preclosure performance in general (i.e., testing and monitoring structures, systems, and components important to safety), it addresses only those aspects of preclosure performance with interactions between engineered and natural systems as might affect

postclosure performance objectives, and for the special case of retrievability. For a construction authorization, the applicant is required to provide a description of a program for performance confirmation that meets the requirements of 10 CFR Part 63, Subpart F.

The Performance Confirmation Program must be explicitly linked to a performance assessment that satisfies 10 CFR 63.113. The Performance Confirmation Program may evolve during construction and operation as the performance assessment is iteratively updated with new information obtained from ongoing performance confirmation monitoring and testing activities. Should a construction authorization be issued, DOE would have appropriate flexibility to change the activities and parameters, as indicated by site and facility conditions and the results from updated performance assessment during construction and operation (NRC, 2001aa, NRC, 1992af).

Information and analyses from the Performance Confirmation Program are required for a license amendment for permanent closure, at the time of which an update is required of the performance assessment of the geologic repository for the period after permanent closure under 10 CFR 63.51(a)(1). The updated assessment for the license amendment for permanent closure must include any performance confirmation data collected under the program required by 10 CFR 63, Subpart F, and pertinent to compliance with 10 CFR 63.113. The option to retrieve waste, which must be maintained until closure, is linked to the completion of the performance confirmation program and update of the performance assessment required for the license amendment for permanent closure in 10 CFR 63.51."
(SER Volume 4, pages 2-1 and 2-2)

"DOE identified in SAR Table 4-1 the natural and engineered barriers that DOE concluded were important to waste isolation. DOE specified the natural and engineered system and components functioning as part of those barriers, and DOE related those barriers to particular performance confirmation activities. In addition to identification of activities linked to barrier performance, DOE identified other activities that supported the ability to retrieve waste, or that confirm disruptive event parameters. DOE identified the Upper Natural Barrier, Engineered Barrier System, and Lower Natural Barrier as important to waste isolation. DOE stated that important barriers are those that prevent or substantially reduce the rate of movement of water or radionuclides from the repository to the accessible environment, or those barriers that prevent the release or substantially reduce the release rate of radionuclides from the waste. For each of the barriers, DOE listed the relevant (i) barrier; (ii) feature, event, or process; (iii) effect on barrier capability; and (iv) core parameter characteristic for each performance confirmation activity in its Performance Confirmation Plan (SNL, 2008aq; addendum to Revision 5, Table A–2[a])."
(SER Volume 4, pages 2-3 and 2-4)

"The DOE Performance Confirmation Plan identified 20 activities for performance confirmation. From SAR Section 4.2 and Table 4-1, these 20 activities, grouped by the SER subsection in which they are evaluated by the NRC staff, are

- SER Section 2.4.3.1.2
    - Precipitation Monitoring
    - Subsurface Water and Rock Testing
    - Unsaturated Zone Testing
    - Saturated Zone Monitoring
    - Saturated Zone Fault Hydrology Testing
    - Saturated Zone Alluvium Testing
- SER Section 2.4.3.2

– Seepage Monitoring

– Drift Inspection

– Thermally Accelerated Drift Near-Field Monitoring

– Thermally Accelerated Drift In-Drift Environment Monitoring

– Subsurface Mapping

– Seismicity Monitoring

– Construction Effects Monitoring

– Thermally Accelerated Drift Thermal-Mechanical Monitoring

- SER Section 2.4.3.3
  – Seal and Backfill Testing

- SER Section 2.4.3.4
  – Dust Buildup Monitoring

  – Waste Package Monitoring

  – Corrosion Testing

  – Corrosion Testing of Thermally Accelerated Drift Samples

  – Waste Form Testing”

  (SER Volume 4, pages 2-4 and 2-5)

The NRC staff reviewed the DOE description of its Performance Confirmation Program in the SAR and additional information describing the Performance Confirmation Program in the Performance Confirmation Plan (SNL, 2008aq) to determine whether the information DOE provided satisfies the requirements for a performance confirmation program in 10 CFR 63.21(c)(17) and 10 CFR Part 63, Subpart F, “Performance Confirmation Program.” The four components of DOE’s performance confirmation program track the four parts of 10 CFR 63, Subpart F; General requirements (10 CFR 63.131); Confirmation of geotechnical and design parameters (10 CFR 63.132); Design testing (10 CFR 63.133); and Monitoring and testing of waste packages (10 CFR 63.134). Based on its review the staff found, with reasonable assurance, that the DOE provided the description required by 10 CFR 63.21(c)(17) that meets the requirements of 10 CFR Part 63, Subpart F. (see SER Volume 4, pages 2-37 to 2-39)

6.4.2 Quality Assurance Program

“DOE’s QA program is described in the Quality Assurance Requirements and Description (QARD) (DOE, 2009gt), which DOE incorporated into the LA by reference. DOE described that applicable requirements will be satisfied primarily through commitments to Quality Assurance Requirements for Nuclear Plants, NQA–1–1983 (American Society of Mechanical Engineers, 1983aa), and other documents whose use, DOE states, the NRC staff finds to be acceptable.” (SER Volume 4, page xvi)

“The purpose of the NRC staff’s review is to determine whether DOE’s QA program description complies with the applicable requirements of 10 CFR Part 63. The NRC staff reviewed DOE’s description of the QA program to be applied to quality-affecting work using guidance in the ‘Yucca Mountain Review Plan’ (YMRP) Section 2.5.1.3 (NRC, 2003aa). YMRP Section 2.5.1.3 identifies the following 18 acceptance criteria that the NRC staff used in its evaluation of DOE’s QA program description:

(1) QA Organization
(2) QA Program
(3) Design Control
(4) Procurement Document Control
(5) Instructions, Procedures, and Drawings
(6) Document Control
(7) Control of Purchased Material, Equipment, and Services
(8) Identification and Control of Materials, Parts, and Components
(9) Control of Special Processes
(10) Inspection
(11) Test Control
(12) Control of Measuring and Test Equipment
(13) Handling, Storage, and Shipping
(14) Inspection, Test, and Operating Status
(15) Nonconforming Materials, Parts, or Components
(16) Corrective Action
(17) Records
(18) Audits" (SER Volume 4, pages 3-1 and 3-2)

The QARD (DOE 2009gt) comprises (i) 18 sections corresponding to the 18 acceptance criteria listed previously; (ii) Supplement I – Software; (iii) Supplement II – Sample Control; (iv) Supplement III – Scientific Investigation; (v) Supplement IV – Field Surveying; (vi) Supplement V – Control of the Electronic Management of Information; (vii) Appendix A – Waste Custodian Interface; (viii) Appendix B (intentionally left blank); and (ix) Appendix C – Storage and Transportation.

> "When DOE submitted the license application, QARD Revision 20 (DOE, 2008af) was in effect. During its initial review of QARD Revision 20, the NRC staff issued to DOE requests for additional information (DOE, 2009gs, 2008aj,ak). Subsequently, DOE incorporated many of its responses to the NRC staff's request for additional information into QARD Revision 21 (DOE, 2009). The NRC staff describes in this SER those responses to NRC staff's requests for information that DOE did not incorporate into QARD Revision 21 (DOE, 2009gt). The NRC staff based its review of DOE's QA program on QARD Revision 21 and the responses to the NRC staff's requests for information not incorporated therein." (SER Volume 4, page 3-3)
>
> "The NRC staff reviewed DOE's QA program description presented in the QARD. Based on its review and evaluation of DOE's QA program description above, the NRC staff finds, with reasonable assurance, that the description required by 10 CFR 63.21(c)(20) adequately addresses how the applicable requirements of 10 CFR 63.142 will be satisfied. The description also adequately addresses the requirements in 10 CFR 63.141, 63.143, and 63.144." (SER Volume 4, page 3-29

6.4.3 Controls to Restrict Access and Regulate Land Uses

> "The regulatory requirements at 10 CFR 63.21(c)(24) require DOE to include in its SAR a description of the controls to restrict access to and regulate land use at the Yucca Mountain site and adjacent areas, including a conceptual design of monuments that would be used to identify the site after permanent closure. The information provided in the SAR must be as complete as possible in light of information that is reasonably available at the time of docketing, in accordance with 10 CFR 63.21(a).
>
> The regulatory requirements at 10 CFR 63.121, regarding ownership and control of interests in land, include provisions for (i) ownership of the land where the geologic repository operations area (GROA) is located and the land being free and clear of significant encumbrances [10 CFR 63.121(a)(1) and (2)]; (ii) additional controls for permanent closure [10 CFR 63.121(b)]; (iii) additional controls through permanent closure [10 CFR 63.121(c)]; and (iv) water rights [10 CFR 63.121(d)(1) and (2)]. In its review, the NRC staff

used applicable guidance in the 'Yucca Mountain Review Plan' (YMRP) Section 2.5.8 (NRC, 2003aa)." (SER Volume 4, page 11-1)

The NRC review identified issues with land ownership and water rights.

Ownership of Land

"The GROA must be located in and on lands that are either acquired lands under the jurisdiction and control of DOE, or lands permanently withdrawn and reserved for its use, as required in 10 CFR 63.121(a)(1). The land on which the GROA will be located must also be free and clear of significant encumbrances, as required by 10 CFR 63.121(a)(2).

In SAR Sections 5.8.1 and 5.8.2.2, DOE provided information regarding land ownership and encumbrances. DOE stated that the GROA and surrounding land within the proposed preclosure controlled area are under the control of several different Federal agencies, including DOE, the U.S. Department of the Interior, and the U.S. Department of Defense. DOE also stated in SAR Section 5.8.1 that it was examining appropriate courses of action that will conform to the requirements of 10 CFR Part 63, including a legislative land withdrawal, to establish effective jurisdiction and control of the land on which the GROA would be located prior to NRC granting a construction authorization. DOE also described, in SAR Section 5.8.1, the course of action it had pursued with respect to ownership of lands. Specifically, in SAR Section 5.8.1, DOE stated that it submitted a land withdrawal bill to Congress in 2007 for the GROA and surrounding area (Senate Bill S.37, introduced May 23, 2007, in the 110th Congress).

In SAR Section 5.8.2.2, DOE stated that the land on which the GROA would be located would be free and clear of encumbrances after completion of the land withdrawal or other acquisition process. In SAR Section 5.8.2.2, DOE stated that the status and occurrence of land encumbrances are dynamic and that a detailed evaluation and discussion of additional land encumbrances are presented in the report, 'Land Records for the Proposed Land Withdrawal Area of the Yucca Mountain Repository' (DOE 2007aa).

The NRC staff evaluated the information DOE provided with respect to land ownership and control, including encumbrances. On the basis of its evaluation, the NRC staff determines that DOE's proposed land withdrawal bill was not enacted into law and that DOE has not completed any other land acquisition process. Therefore, the NRC staff concludes that DOE neither has acquired lands to be under its jurisdiction and control for the GROA, nor have the lands for the GROA been permanently withdrawn and reserved for DOE's use. In addition, because DOE has not completed a land withdrawal or other acquisition process, DOE has not demonstrated that such land would be free and clear of significant encumbrances. Therefore, the NRC staff finds that the requirements in 10 CFR 63.121(a)(1) and (a)(2) for ownership and control of the GROA are not satisfied." (SER Volume 4, page 11-2)

Water Rights

"DOE must obtain such water rights as may be needed to accomplish the purpose of the GROA, as required in 10 CFR 63.121(d)(1). In SAR Section 5.8.4 and DOE's response to the NRC staff's RAI (DOE, 2009au), DOE described its approach for obtaining such water rights. DOE estimated a maximum annual water demand of 53.0 hectare-meters [430 acre-ft] for construction (prior to receipt and possession of waste) and a maximum annual water demand of 40.7 hectare-meters [330 acre-ft] for operations (after receipt and possession waste). In SAR Section 5.8.4, DOE stated that it filed a water appropriation request with the Nevada State Engineer on July 22, 1997, for the permanent rights to 53.0 hectare-meters [430 acre-ft] annually from five wells within the proposed preclosure controlled area for the purpose of constructing and operating the repository. In SAR Section 5.8.4, DOE stated that the Nevada State Engineer denied the DOE water appropriation permit applications and that the U.S. Department of Justice, on behalf of DOE, appealed this decision.

The NRC staff evaluated the information DOE provided on obtaining water rights as may be needed to accomplish the purpose of the GROA. DOE's actions to obtain water rights for this purpose have not been successful. Therefore, the NRC staff finds that the regulatory requirement in 10 CFR 63.121(d)(1) has not been satisfied, because DOE has not obtained such water rights that DOE determined may be needed to accomplish the purpose of the GROA." (SER Volume 4, page 11-6)

6.4.4 Conclusions

"SER Volume 4 documents the results of the NRC staff's evaluation to determine whether DOE's research and development program, performance confirmation program, and other programmatic and administrative controls, systems, and programs meet applicable regulatory requirements. Based on its review, the NRC staff finds, with reasonable assurance, that, except as noted below, DOE has addressed applicable requirements including 10 CFR 63.21, 'Content of Application'; 10 CFR 63.121, 'Land Ownership and Control'; 10 CFR Part 63, Subpart D, 'Records, Reports, Tests, and Inspections'; 10 CFR Part 63, Subpart F, 'Performance Confirmation Program'; 10 CFR Part 63, Subpart G, 'Quality Assurance'; 10 CFR Part 63, Subpart H, 'Training and Certification of Personnel'; and 10 CFR Part 63, Subpart I, 'Emergency Planning Criteria.'

The NRC staff is proposing one condition of construction authorization in this SER Volume related to the description of programs designed to resolve safety questions. Pursuant to 10 CFR 63.32(b)(4), in the event that DOE identifies any safety questions that would require research and development programs in the future, the results of those programs must be appropriately reported to the NRC.

The NRC staff finds that DOE has not met the requirements 10 CFR 63.121(a) and 10 CFR 63.121(d)(1) regarding ownership of land and water rights, respectively." (SER Volume 4, page vii)

6.4.5 References

DOE. 2009au. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 5.8), Safety Evaluation Report Vol. 4, Chapter 2.5.8." Letter (May 6) J.R. Williams to F. Jacobs (NRC). ML091330698. Washington, DC: DOE, Office of Technical Management.

DOE. 2009gs. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 5.1), Safety Evaluation Report Vol. 4, Chapter 2.5.1, Set 1." Letter (August 11) J.R. Williams to B.J. Benney (NRC). ML092360006. Washington, DC: DOE, Office of Technical Management

DOE. 2009gt. DOE/RW–0333P, "Quality Assurance Requirements and Description (QARD)." Rev. 21. ML090230236. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2008af. DOE/RW–0333P, "Quality Assurance Requirements and Description (QARD)." Rev. 20. ML080450334. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2008aj. “Yucca Mountain—Request for Additional Information Regarding License Application (Safety Analysis Report Section 5.1), Safety Evaluation Report Vol. 4, Chapter 2.5.1, Set 1.” Letter (December 2) J.R. Williams to B.J. Benney (NRC). ML083380568. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ak. “Yucca Mountain—Request for Additional Information Regarding License Application (Safety Analysis Report Section 5.1), Safety Evaluation Report Vol. 4, Chapter 2.5.1, Set 1.” Letter (December 10) J.R. Williams to B.J. Benney (NRC). ML090080758. Washington, DC: DOE, Office of Technical Management.

DOE. 2007aa. “Land Records for the Proposed Land Withdrawal Area of the Yucca Mountain Repository.” TDR–MGR–ND–000003. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

NRC. 2003aa. NUREG–1804, “Yucca Mountain Review Plan—Final Report.” Rev. 2. ML032030389. Washington, DC: U.S. Nuclear Regulatory Commission.

NRC. 2001aa. “Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, NV: Final Rule.” Federal Register. Vol. 66, No. 213. Pp. 55,732–55,816. Washington, DC: U.S. Nuclear Regulatory Commission.

NRC. 1992af. NUREG-0804, “Staff Analysis of Public Comments on Proposed Rule 10 CFR Part 60 Disposal of High-Level Radioactive Wastes in Geologic Repositories.” Washington, DC: U.S. Nuclear Regulatory Commission.

SNL. 2008aq. “Performance Confirmation Plan.” TDR–PCS–SE–000001. Rev. 05. ADD 01, CAN 01. ML090770723, ML090770724. Las Vegas, Nevada: Sandia National Laboratories.

6.5 License Conditions (NUREG-1949 – SER Volume 5)

> “Volume 5 of this Safety Evaluation Report (SER) is entitled ‘Proposed Conditions on the Construction Authorization and Probable Subjects of License Specifications.’ Although the Yucca Mountain Review Plan uses the heading “License Specifications” for this volume, the title for Volume 5 was revised to more accurately reflect the contents of the volume. Volume 5 includes information and findings from the other four volumes of the SER that document the NRC staff’s review of the SAR DOE provided in its June 3, 2008, license application submittal (DOE, 2008ab), as updated on February 19, 2009 (DOE, 2009av). The NRC staff also reviewed information DOE provided in response to the NRC staff’s requests for additional information (RAIs) and other information that DOE provided related to the SAR. In particular, this SER Volume 5 documents the NRC staff’s proposed conditions of construction authorization, including proposed conditions documented in other SER Volumes, and review of DOE’s probable subjects of license specifications.” (SER Volume 5, page xiii)

6.5.1 Proposed Conditions on the Construction Authorization

The proposed conditions were based on the conditions of the Construction Authorization (CA) as required 10 CFR 63.32 and the Nuclear Waste Policy Act (NWPA), as amended, as well as proposed conditions identified in other volumes of the NRC staff’s SER.

<u>Requirements of 10 CFR 63.32(c)</u>

> "The requirements in 10 CFR 63.32(c) specify that the Commission include in a CA restrictions on subsequent changes to the features of the geologic repository and the procedures authorized. These restrictions can include measures to prevent adverse effects on the geologic setting as well as measures related to the design and construction of the GROA for which there are three distinct categories of descending importance to public health and safety. The categories in 10 CFR 63.32(c)(1)–(3) provide for differing levels of restrictions.
>
> In Chapter 5 of the SAR, DOE states that it is committed to apply, after issuance of a CA, any specific conditions imposed in accordance with 10 CFR 63.32 to any changes to the repository design or procedures as described in the SAR." (SER Volume 5, page 1-5)

Based on its review concerning restrictions on subsequent changes to the features of the geologic repository and the procedures authorized, the NRC staff proposed the following conditions be included in a CA issued by the Commission:

> "Pursuant to 10 CFR 63.32(c), the licensee is restricted from making any changes, without 60 days prior notice to the Commission and prior Commission approval, that (i) require an amendment of the construction authorization pursuant to the criteria in 10 CFR 63.44(b)(2); or (ii) change land controls for the geologic setting of the repository related to compliance with the preclosure performance objectives [10 CFR 63.111(a) and (b)], emergency planning (10 CFR 63.21(c)(21), 10 CFR 63.161), and controls to prevent adverse human actions that could significantly reduce the geologic repository's ability to achieve isolation [10 CFR 63.121(b)].
>
> The licensee is restricted from making any changes to the scope (including the frequency of monitoring and maintenance activities) of the monitoring and maintenance programs for ensuring the stability of repository drifts, as described in SAR Section 1.3.1.2.1.6, without 60 days prior notice to the Commission. In this notice, the applicant should confirm any proposed change will not adversely impact the reliability or safety functions for the potentially impacted SSCs important to safety or barriers important to waste isolation. Changes to the scope of the monitoring and maintenance programs for ensuring stability of repository drifts may not be changed without prior Commission approval if, after receiving the required 60 day notice, the Commission so orders." (SER Volume 5, page 1-8)

Conditions on the Construction Authorization Based on Technical Review, Part 63 Requirements and Statutory Direction

> "In addition to the proposed conditions discussed in SER Section 2.5.10.1.3.1.1.2, the NRC staff identified conditions on the construction authorization based on its review documented in SER Volumes 1–4, the regulatory requirements in 10 CFR 63.32, and statutory requirements. In its reviews of General Information (SER Volume 1) and Postclosure Safety (SER Volume 3), the NRC staff did not identify any conditions for a construction authorization. The NRC staff did identify proposed conditions of construction authorization in its evaluations of Preclosure Safety (SER Volume 2) and Administrative and Programmatic Requirements (SER Volume 4). Table 2.5-1 provides the proposed conditions and, as appropriate, the SER Section where the proposed condition is discussed." (SER Volume 5, page 1-8)
>
> "As noted previously, the NRC staff determined that DOE has not satisfied certain regulatory requirements. The NRC staff's proposed conditions, based on its review of the SAR, RAI responses, and supporting information, do not represent an approach for addressing regulatory requirements that DOE has not met regarding ownership and control of certain land and water rights. Should the applicant provide additional information, NRC staff may remove or revise a condition, or could add one or more conditions, based on its review of the information." (SER Volume 5, page 1-9)

6.5.2 Probable Subjects for License Specifications

"NRC regulations at 10 CFR 63.21(c)(18) require the U.S. Department of Energy (DOE) to provide, as part of the Safety Analysis Report (SAR), probable subjects of license specifications. By letter dated June 3, 2008, as supplemented on February 19, 2009 (DOE 2009av), the DOE provided in its license application [SAR Volume 5, Section 5.10 (DOE, 2008ab)] its proposals for potential subjects of license specifications." (SER Volume 5, page 2-1)

"DOE has identified the following as probable subjects for license specifications:

(1) probable subjects of license specifications for operation (DOE 2009av, Table 5.10-1)
- (a) surface ITS confinement HVAC systems
- (b) ITS power system (e.g., ITS direct current power and diesel generators)
- (c) ITS HVAC system supporting cooling of ITS electrical and control equipment
- (d) ITS fire detection and suppression system
- (e) TAD canister dewatering and drying
- (f) wet handling facility pool boron concentration
- (g) ITS radiation detectors and interlocks

(2) probable subjects of license specifications for design features (DOE 2009av, Table 5.10-2)
- (a) repository location (e.g., site boundaries)
- (b) geologic constraints for emplacement drifts (e.g., depth above groundwater)
- (c) location, size and capacity of aging pads
- (d) waste form limits (e.g., maximum burnup, enrichment, and time out of reactor)
- (e) waste package limits (e.g., waste package configuration)
- (f) drip shield limits (e.g., interlocking design features)

(3) probable subjects of license specifications for administrative controls (DOE 2009av, Table 5.10-3)
- (a) responsibilities (e.g., site operations manager, waste handling manager)
- (b) organization (e.g., organization charts, functional descriptions of departmental responsibilities and relationships)
- (c) repository staff qualifications (e.g., operation staff be trained and certified)
- (d) procedures (e.g., emergency operations, alarms and annunciators, maintenance)
- (e) high radiation areas (e.g., alternative methods to control access)
- (f) license specifications bases control program

(4) probable subjects of license specifications for administrative controls for programs/manuals unique to the operation of a geologic repository and GROA required to ensure operations are consistent with the assumptions of the PCSA or postclosure analyses (DOE 2009av, Table 5.10-3)
- (a) waste form and waste package qualification program
- (b) canister and transportation cask acceptance program
- (c) reliability centered maintenance
- (d) waste package loading, handling, and emplacement program
- (e) subsurface committed materials control program
- (f) access control program (control access outside the GROA to avoid disturbance of site)
- (g) fire protection program (e.g., ignition source control, fire barriers)
- (h) technical requirements manual (e.g., approval process for changes to Technical Requirements Manual and associated bases)

DOE stated that (i) the limiting conditions for operation will include specific surveillance testing requirements or other inspections to verify that process variables are maintained within proper ranges or to support determinations of SSC capability to function in a manner that bounds the nuclear safety design bases for the PCSA and the postclosure performance assessment; (ii) the configuration management system will include necessary reviews to ensure compliance with 10 CFR 63.44 for proposed changes to the SAR that could impact the repository design, analysis, or operation; and (iii) it will submit a proposed draft set of license specifications to the NRC prior to issuance of a license to receive and possess and the final license specifications issued by the NRC are expected to be incorporated as an appendix to the license to receive and possess." (SER Volume 5, pages 2-3 to 2-5)

"Based on this evaluation, the NRC staff concludes, with reasonable assurance, that the requirements of 10 CFR 63.21(c)(18) are satisfied because (i) the applicant's identification and technical justification of the probable subjects for license specifications are acceptable; and (ii) the applicant acceptably described its plans for implementation of the probable subjects of license specifications." (SER Volume 5, page 2-7)

6.5.3 References

DOE. 2009ab. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 2." Letter (February 23) J.R. Williams to J.H. Sulima (NRC). ML090550101, ML090550099. Washington, DC: U.S. Department of Energy, Office of Technical Management.

DOE. 2009av. DOE/RW–0573, "Update to the Yucca Mountain Repository License Application (LA) for Construction Authorization." Rev. 1. February 2009. ML090700817, ML090710096. Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

6.6 SER Conclusions

"The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed and evaluated the U.S. Department of Energy's (DOE) Safety Analysis Report (SAR), provided in its June 3, 2008, license application (LA), as updated on February 19, 2009. The NRC staff also reviewed information DOE provided in response to the NRC staff's requests for additional information and other information that DOE provided related to the SAR. The staff has documented the results of its review in its Safety Evaluation Reports (SER) Volumes 1 through 5. In summary, the NRC staff has found that

- DOE has adequately described the proposed geologic repository at Yucca Mountain including the information, analyses, and programs associated with the preclosure and postclosure performance of the repository as specified in 10 CFR 63.21 of NRC's regulations.

- DOE has adequately described (i) the material control and accounting program; and (ii) security measures for physical protection in accordance with 10 CFR 73.51 (SER Volume 1: General Information).

- The NRC staff has found, with reasonable assurance, that subject to proposed conditions of the construction authorization, DOE's design of the proposed geologic repository operations area (GROA) and preclosure safety analysis complies with the preclosure performance objectives at 10 CFR 63.111 and the requirements for preclosure safety analysis of the GROA at 10 CFR 63.112. (SER Volume 2: Repository Safety Before Permanent Closure).

- The NRC staff has found, with reasonable expectation, that the proposed Yucca Mountain repository design meets the applicable postclosure performance objectives in Subpart E, including the requirement that the repository be composed of multiple barriers; and (ii) based on performance assessment evaluations that are in compliance with applicable regulatory requirements, meets the 10 CFR Part 63, Subpart L limits for individual protection, human intrusion, and separate standards for protection of groundwater. (SER Volume 3: Repository Safety After Permanent Closure).

- NRC staff has found, with reasonable assurance, that, except as noted below, DOE has addressed applicable administrative and programmatic requirements regarding, “Land Ownership and Control”; “Records, Reports, Tests, and Inspections”; “Performance Confirmation Program”; “Quality Assurance”; “Training and Certification of Personnel”; and “Emergency Planning Criteria.” The NRC staff finds that DOE has not met the requirements in 10 CFR 63.121(a) and 10 CFR 63.121(d)(1) regarding ownership of land and water rights, respectively. (SER Volume 4: Administrative and Programmatic Requirements.

- The NRC staff has found, with reasonable assurance, that the requirements of 10 CFR Part 63.21(c)(18) are satisfied because: (i) the applicant’s identification and technical justification of the probable subjects for license specifications are acceptable; and (ii) the applicant acceptably described its plans for implementation of the probable subjects of license specifications. (SER Volume 5: Proposed Conditions on the Construction Authorization and Probable Subjects of License Specifications)

As noted above, the NRC staff determined that DOE has not satisfied certain regulatory requirements regarding ownership of the land where the GROA is located and water rights. In addition, a supplement to DOE’s environmental impact statement has not yet been completed. Thus, the NRC staff is not recommending issuance of a construction authorization at this time.

Nevertheless, in accordance with 10 CFR Part 63 requirements, SER Volume 5 identifies conditions of Construction Authorization proposed by the NRC staff based on its review of DOE’s SAR, supplemental documents referenced in the SAR, and DOE’s responses to NRC staff requests for additional information (RAIs). These NRC staff proposed conditions could be included in a Construction Authorization if there is a Commission decision to authorize construction. However, these proposed conditions do not represent an approach for addressing the regulatory requirements regarding ownership of the land and water rights that DOE did not meet. Should the applicant provide additional information, the NRC staff may remove or revise a condition stated here or could add one or more conditions, based on its review of that information.” (SER Volume 5, pages 3-1 and 3-2)

7.0 Environmental Impact Statement

“As described in the NWPA, Section 114(f), DOE prepared a final environmental impact statement (EIS) in February 2002 to meet National Environmental Policy Act (NEPA) requirements related to the construction, operation, and closure of a potential geologic repository for high-level radioactive waste at Yucca Mountain, Nevada (DOE, 2002). This EIS accompanied the Secretary of Energy’s site recommendation to the President on February 14, 2002. In July 2002, Congress passed and the President signed a joint resolution designating Yucca Mountain as the site for development as a geologic repository.

In October 2006, DOE published a notice of intent in the Federal Register (DOE, 2006a) to prepare a supplemental EIS to update the 2002 EIS (Repository Supplemental EIS). At the same time, DOE announced its intent to develop an EIS for the Nevada rail alignment (Rail Alignment EIS) and a supplement to the rail corridor analyses presented in the 2002 EIS [Rail Corridor Supplemental EIS (SEIS)] (DOE, 2006b). The draft EIS and draft supplemental EISs were issued for public comment on October 12, 2007 (DOE, 2007a, b, c). The public comment period for the draft EIS and supplemental EISs ended on January 10, 2008, and DOE published the final supplemental EISs and the final Rail Alignment EIS on June 16, 2008 (DOE, 2008b, c, d).

DOE submitted the 2002 EIS with the license application on June 3, 2008. DOE submitted the Repository Supplemental EIS on June 16, 2008, in accordance with 10 CFR § 51.67(b). The Rail Corridor SEIS and Rail Alignment EIS were also provided on June 16, 2008. In accordance with NWPA, Section 114(f), NRC is to adopt the DOE EIS to ‘the extent practicable.’ As described in NRC NEPA-implementing regulations in 10 CFR § 51.109(a)(1), the EIS is considered to include ‘…any supplement thereto.” The regulations for the NRC adoption determination are set forth in 10 CFR § 51.109(c). These regulations state that the NRC staff “…will find that it is practicable to adopt any environmental impact statement prepared by the Secretary of Energy in connection with a geologic repository proposed to be constructed under Title I of the NWPA of 1982, as amended, unless:

- The action proposed to be taken by the Commission differs from the action proposed in the license application submitted by the Secretary of Energy; and the difference may significantly affect the quality of the human environment; or
- Significant and substantial new information or new considerations render such environmental impact statement inadequate.’

Using these criteria, NRC may adopt the EIS and any supplements, adopt them in part indicating a supplement is needed in part, or not adopt them, requiring supplementation.”
(NRC 2008; pages 1-1 and 1-2)

7.1 NRC’s EIS Adoption Determination

“Consistent with the NWPA’s intention to eliminate duplication and with NRC’s regulations in 10 CFR § 51.109, the NRC staff’s adoption review is neither a duplication of DOE’s efforts nor a detailed review of all technical aspects of the analyses contained in these EISs. Further, an NRC staff determination of adoption of these EISs does not necessarily mean that NRC would have independently arrived at the same conclusions as DOE on matters of fact or policy. The staff recognizes that DOE, as the lead agency for implementing the NEPA process for the proposed repository, may reach conclusions that are different from those others might make.

Consistent with NUREG–1748 (“Environmental Review Guidance for Licensing Actions Associated with NMSS Programs”), the NRC staff considers that the use of a regulatory requirement to limit an analysis of impacts is not necessarily appropriate in the context of NEPA. As discussed further in Section 3.2.1.4.2, the NRC staff concludes that the discussion regarding the environmental impacts on groundwater requires further supplementation." (NRC 2008, page 1-3)

After its review, the NRC staff concluded:

"… that the 2002 EIS, the Repository Supplemental EIS, and the Rail Corridor SEIS meet NRC completeness and adequacy requirements in 10 CFR § 51.91 and in 10 CFR Part 51, Subpart A, Appendix A, and that the EISs are generally consistent with NRC's NEPA guidance in NUREG–1748 (NRC, 2003a). The NRC staff has determined that significant and substantial new considerations related to groundwater analyses in the 2002 EIS (DOE, 2002) and in the Repository Supplemental EIS (DOE, 2008b) render those analyses of the EISs inadequate without further supplementation. These considerations are addressed in depth in Section 3.2.1.4.2 of this report. The staff has not identified other significant or substantial new information or considerations that would render the EISs (DOE, 2008b, c; 2002) inadequate."
(NRC 2008, page 3-15)

In particular, the NRC stated that:

"A supplement should include the following additional information:

- A description of the locations of potential natural discharge of contaminated groundwater for present and expected future wetter periods (for example, as discussed in DOE, 2008a, Safety Analysis Report, Section 2.3.1.2).

- A description of the physical processes at the surface discharge locations that can affect accumulation, concentration, and potential remobilization of groundwater-borne contaminants.

- Estimates of the amount of contaminants that could be deposited at or near the surface. This involves estimates of the amount of groundwater involved in discharge or near-surface evaporation, the amounts of radiological and non-radiological contaminants in that water, contaminant concentrations in the resulting deposits, and potential environmental impacts (e.g., effects on biota)."
(NRC 2008, page 3-12)

7.2 NRC's EIS Supplement

Based on the outcome of the NRC staff's Adoption Determination Review (ADR), the Commission requested that DOE complete an EIS supplement. DOE declined to prepare the supplement and, following publication of SER Volumes 1-5, the Commission directed the NRC staff to develop the supplement. The NRC staff announced its intent to develop this supplement in the Federal Register (FR) on March 12, 2015 (80 FR 13029). The NRC staff also issued a press release and notified the hearing participants and other stakeholders. Pursuant to 10 CFR 51.26(d), the NRC staff did not conduct scoping for this supplement, the scope of which was established by the ADR. The NRC staff did not identify any cooperating agencies for this supplement, nor did the NRC staff receive any formal requests for cooperating agency status. The NRC staff provided a 60-day public comment period for this draft supplement that was later extended to 91 days. During the comment period, the NRC staff conducted five public meetings, which included both remote (video and teleconference) meetings and in-person meetings held in Las Vegas, Nevada, and Amargosa Valley, Nevada. The NRC received over 1,200 oral and written comments on the draft supplement. The NRC Staff published its Final supplement to DOE's EIS, titled, "Supplement to the U.S. Department of Energy's Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada" on May 2016 (NRC 2016 – NUREG-2184).

"In Section 3.2.1.4.2 of the Adoption Determination Report (ADR), the NRC staff found that DOE's environmental impact statements (EISs) did not adequately characterize impacts from potential contaminant releases to groundwater and from surface discharges of groundwater. Specifically, DOE's analysis does not provide adequate discussion of the cumulative amounts of radiological and nonradiological contaminants that may enter the groundwater over time and how these contaminants would behave in the aquifer and surrounding environments. This supplement provides the information the NRC staff identified as necessary in its ADR. Two distinct but related aspects of potential impacts on the groundwater system are addressed in this supplement. These are (i) the nature and extent of the repository's impacts on groundwater in the aquifer (beyond the postclosure compliance location) and (ii) the potential impacts of the discharge of potentially contaminated groundwater to the ground surface.

This supplement describes the affected environment with respect to the groundwater flow path for potential contaminant releases from the repository that could be transported beyond the postclosure compliance location through the volcanic-alluvial aquifer in Fortymile Wash and the Amargosa Desert, and to the Furnace Creek/Middle Basin area of Death Valley. … Thus, this supplement provides a description of the flow path from the postclosure compliance location to Death Valley, the locations of current groundwater withdrawal, and locations of potential natural discharge along the groundwater flow path. The supplement evaluates the potential groundwater-related environmental impacts at these locations over a one-million year period following repository closure." (NRC 2016, page xi)

"This supplement describes the potential impacts that could occur under different climate conditions and under different assumptions for groundwater withdrawal. The analysis in this supplement encompasses the range of credible future climates and human activities affecting groundwater in the Yucca Mountain region, and includes conservative assumptions for future conditions and processes. Future climates are projected to include periods that are relatively hot and dry (similar to present-day conditions) and periods that are relatively cooler and wetter over the one-million-year time period." (NRC 2016, page xi)

NRC's review considered the significance categories for potential environmental impacts "based on NRC guidance (NRC, 2003) and are characterized as follows:

SMALL—The environmental impacts are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource.

MODERATE—The environmental impacts are sufficient to alter noticeably, but not to destabilize, important attributes of the resource.

LARGE—The environmental impacts are clearly noticeable and are sufficient to destabilize important attributes of the resource.

This NRC staff supplement evaluates the direct, indirect, and cumulative impacts on water and soil, public health, ecology, historic and cultural resources, and environmental justice for locations beyond the postclosure compliance location. The locations of the affected environment are described in Chapter 2, which include potential locations for groundwater pumping and natural surface discharge beyond the postclosure compliance location downstream along the groundwater flow path to Death Valley.

The NRC staff finds that all of the impacts on the resources evaluated in this supplement are SMALL. The NRC staff's analysis includes the impact of potential radiological and nonradiological releases from the repository on the aquifer and at surface discharge locations of groundwater beyond the postclosure compliance location. The peak annual individual radiological dose at any of the evaluated locations is 1.3 mrem [0.013 mSv] from pumping and irrigation at the Amargosa Farms area. The NRC staff concludes that all estimated radiological doses are SMALL because they are a small fraction of background radiation dose of 300 mrem/yr [3 mSv/yr] (including radon), and much lower than the NRC annual dose standards for a Yucca Mountain repository in title 10 of Code of Federal Regulations (CFR) Part 63 {15 mrem [0.15 mSv] for the first 10,000 years, and 100 mrem [1 mSv] for one million years, after permanent closure}. The NRC staff's peak dose estimates accounted for uncertainty in climate and in groundwater pumping at the Amargosa Farms area. Based on conservative assumptions about the potential for health effects from exposure to low doses of radiation, the NRC staff expects that the estimated radiation dose would contribute only a negligible increase in the risk of cancer or severe hereditary effects in the potentially exposed population. Impacts to other resources at all of the affected environments beyond the postclosure compliance location from radiological and nonradiological (i.e., chemical) material from the repository would also be SMALL, based on low estimated levels of the evaluated constituents in those potentially affected areas.

The cumulative impact analysis in Chapter 4 of this supplement contains the NRC staff's evaluation of the cumulative impacts for direct and indirect impacts identified in Chapter 3 when aggregated with the impacts of other actions that could affect the same resources. The NRC staff also evaluates how its findings in Chapter 3 and cumulative impact findings in Chapter 4 affect the conclusions provided by DOE in its assessment of cumulative impacts on groundwater in Chapter 8 of its EIS (DOE, 2002) and Chapter 8 of its SEIS (DOE, 2008b[5])." (NRC 2016, pages 5-1 and 5-2)

"The cumulative impact analysis concludes that, when considered in addition to the incremental impacts of the proposed action, the potential impacts of other past, present, or reasonably foreseeable future actions would also be SMALL." (NRC 2016, page xii)

[5] Note the cited text uses DOE 2008a as the reference for the SEIS, however in this document the reference for the SEIS is DOE 2008b – to avoid confusion in this document the citation was changed to DOE 2008b.

7.3 References

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Appendix A: Development of Regulations for Geologic Disposal of High-Level Waste and Spent Nuclear Fuel

Appendix A is at ADAMS No. ML21251A620

Appendix B: Postclosure Safety Review for a Potential Repository at Yucca Mountain

Appendix B is at ADAMS No. ML21259A156