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Yucca Mountain Project

PRECLOSURE SEISMIC DESIGN AND PERFORMANCE DEMONSTRATION METHODOLOGY FOR A GEOLOGIC REPOSITORY AT YUCCA MOUNTAIN TOPICAL REPORT

YMP/TR-003-NP

Revision 5

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U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Las Vegas, Nevada

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CHANGE HISTORY

Revision Number	**Interim Change No.**	**Effective Date**	**Description of Change**
00	00	10/24/1995	Initial issue. The report [MOL.19960404.0325] was transmitted to NRC on 10/31/95 [letter: MOL.19960404.0324]. The Management and Operating Contractor deliverable to DOE resulting in this report is identified as BC0000000-01717-5705-0010 [MOL.19950913.0231].
01	00	10/10/1996	Complete revision of document and methodology to comply with requirements of NRC-proposed rulemaking for 10 CFR Part 60 design basis events (i.e., 60 FR 15180). Terminology of seismic safety performance categories (PC-1 to PC-4) replaced by seismic frequency categories (FC-1 and FC-2). Revision 01 also addressed comments by NRC on revision 0. This report [MOL.19970114.0027; MOL.19970402.0168] was transmitted to NRC on 10/25/96 [MOL.19970114.0026].
02	00	08/ 07/1997	Revision of approach to address final NRC rulemaking for 10 CFR Part 60 design basis events (i.e., 61 FR 64257). Revision also includes changes to mitigation of fault displacement hazards in Section 4, with increased discussion of criteria for fault avoidance. Text was also updated for (final) amendments to 10 CFR Part 50 and 10 CFR Part 100 (61 FR 65157). This report [MOL.19971009.0412; MOL.19971009.0413] was transmitted to NRC on 8/27/97 [MOL.19971017.0593].
03	00	10/2004	Complete revision to comply with 10 CFR Part 63 (replacing 10 CFR Part 60) and incorporating information from the *Yucca Mountain Review Plan*. In addition, the revision incorporates regulatory precedent and decision-making that have occurred since revision 2 as well as recent seismic design experience. Terminology of seismic frequency categories (FC-1 and FC-2) is replaced with DBGM levels (DBGM-1 and DBGM-2) and design basis fault displacement (DBFD-1 and DBFD-2). This report [MOL.20041103.0002] was transmitted to NRC on 11/9/04 [MOL.20050310.0275].

Revision Number	Interim Change No.	Effective Date	Description of Change
			Two BSC controlled documents were developed in support of YMP/TR-003-NP, Revision 3. They are: TDR-WHS-MD-000004, Revision 0, *Preclosure Seismic Design Methodology For A Geologic Repository At Yucca Mountain* (DC# 36953) [DOC.20040121.0008] on1/22/04; and TDR-WHS-MD-000004, Revision 1, *Preclosure Seismic Design Methodology For A Geologic Repository At Yucca Mountain* (DC#42547) [DOC.20040827.0011; MOL.20041111.0222] on 8/27/04, which superseded the initial version.
04	00	11/2006	Extensive revision to update and supplement discussion of regulatory context for risk-informed DBGM levels. Also, revisions respond to the January 24, 2006, letter from NRC transmitting comments on YMP/TR-003-NP, Revision 3 (Kokajko 2006) [MOL.20060313.0183]. These revisions include probabilistic seismic analyses to demonstrate compliance with the preclosure performance objectives in 10 CFR 63.111. For this revision, OCRWM elected to have the report developed by BSC and subsequent to DOE acceptance and approval, submitted to the NRC. The document is to be submitted for BSC document control in accordance with LP-6.3Q-BSC, *Document Control*. Revision 4 [DOC.20061205.0004] supersedes Topical Report YMP/TR-003-NP, Revision 3. [Note that predecessors of this current version have not been controlled by BSC and, therefore, the DOE predecessors of this document are not in BSC's Controlled Documents Information System.] In addition, Revision 4 also supersedes two BSC documents developed in support of YMP/TR-003-NP, Revision 3, specifically, TDR-WHS-MD-000004, Revisions 0 and 1 [included in the Controlled Documents Information System], identified above in the change history for Revision 3.
05	00	6/2007	Several changes made for clarification in response to review by DOE. Clarifications include emphasis on

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			probabilistic compliance demonstration, reference to the most recent applicable documents and NRC guidance documents, consistency with recent revisions to NUREG-0800, and consistency with other aspects of the preclosure safety analysis.

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1. INTRODUCTION

This topical report describes the methodology and criteria that the U.S. Department of Energy (DOE) intends to use for the preclosure seismic design of structures, systems, and components (SSCs) that are important to safety (ITS) in the geologic repository operations area both on the surface and in the subsurface. 10 CFR Part 63 states that for a license to be issued for the operation of a high-level radioactive waste (HLW) repository, the U.S. Nuclear Regulatory Commission (NRC) must find that the facility will not constitute an unreasonable risk to the health and safety of the public (10 CFR 63.41[c]). 10 CFR 63.21(c)(5) requires that a preclosure safety analysis (PCSA) be performed to ensure preclosure performance objectives (10 CFR 63.111) have been met. The PCSA is a systematic examination of the site, design, and potential hazards (10 CFR 63.102[f]), including a comprehensive identification of potential event sequences. Potential naturally occurring hazards include those event sequences initiated by earthquake ground motions or fault displacements due to earthquakes.

In accordance with 10 CFR 63.2, design bases for the repository include consideration of severe natural events, such as earthquakes. The preclosure performance objectives for the geologic repository operations area are given in 10 CFR 63.111 and it is required that the license application (LA) show the relation between design criteria and meeting the preclosure performance objectives (10 CFR 63.21[c][3][ii]). The measure of acceptable risk is expressed in terms of allowable consequences for Category 1 or Category 2 event sequences. Allowable consequences are given as performance objectives (i.e., dose limits) in 10 CFR 63.111.

The PCSA must also include a discussion of the design and how design criteria are related to design bases such that compliance with the preclosure performance objectives is ensured (10 CFR 63.112[f]). This topical report responds to 10 CFR Part 63 requirements with respect to preclosure seismic design, describes the seismic design methodology that the DOE intends to use, and defines a methodology that will provide a basis for NRC to find reasonable assurance that the preclosure performance objectives contained in 10 CFR 63.111 are achieved.

This revision to the topical report supersedes *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (DOE 2004) and has been updated to reflect consideration of recent seismic design experience and regulatory interpretations that have occurred during the past several years. These activities provide a context and basis for the seismic design methodologies put forward in this document.

10 CFR Part 63 does not prescribe a specific approach to developing seismic design bases. Rather, the regulation is risk-informed and performance-based, which means that the demonstration of compliance with the preclosure performance objectives is the ultimate goal to be used in the establishment of design bases. Therefore, the DOE has developed a preclosure seismic design methodology that consists of two parts: (1) seismic design criteria, including design basis ground motions (DBGM) and codes, standards, and acceptance criteria that are consistent with applicable regulatory precedents from commercial nuclear licensing, and (2) a compliance demonstration that shows that the preclosure performance objectives in 10 CFR 63.111(a), (b), and (c) have been met. To do so, this preclosure seismic design methodology is integrated with PCSA, and both design methodology and safety analyses are used to demonstrate compliance.

This document provides the methodology for preclosure seismic design, which includes the establishment of the seismic DBGM and fault displacement hazard levels for ITS SSCs. ITS SSCs are credited with preventing or mitigating the consequences of seismically initiated event sequences. The methodology includes:

- A comprehensive and systematic identification of seismically-initiated event sequences and categorization of ITS SSCs according to their potential to prevent or mitigate event sequences

- An analysis of the potential radiological consequences of seismically-initiated event sequences and assignment of DBGM levels

- Use of well established seismic design and analysis methods for the design of structures that have nuclear facility precedent, including the use of the codes and standards identified in NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* (NRC 1987[1]), for the design of structures

- Evaluation of seismic margins of specific SSCs to ensure that the combination of DBGMs and design procedures are adequately conservative

- Probabilistic seismic analyses to assess the probability of seismically initiated event sequences and to demonstrate compliance with 10 CFR 63.111.

1.1 BACKGROUND

Seismic design methodology for the Yucca Mountain Project (YMP) was originally documented in the second of two topical reports on seismic hazards and preclosure seismic design of the planned geologic repository at Yucca Mountain, Nevada. A third topical report was originally planned but the information was subsequently documented in other technical reports as discussed later in this section. The first seismic topical report (STR#1), *Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain* (YMP 1997a), described the methodology to be used to evaluate the vibratory ground motion and fault displacement seismic hazard at Yucca Mountain. The topical report was reviewed by the NRC staff and, after comment resolution, concluded that there were no further questions related to it, pending review of the three proposed topical reports. Subsequently, the seismic hazard methodology was implemented in *Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada* (CRWMS M&O 1998) and the results of that probabilistic seismic hazard analysis (PSHA) provide a basis for subsequent seismic design inputs, for use in both preclosure design and postclosure performance assessments.

The initial issue and first revision of the second seismic topical report (STR#2), *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1997b), described the preclosure seismic design methodology to be used for ITS SSCs. STR#2 described criteria and procedures for determining design basis vibratory ground motions in terms of the

[1] Citation to NUREG-0800 in this report is to the original 1987 publication, but reference will be made to specific approved sections issued subsequent to 1987, as applicable.

mean annual exceedance probabilities for ITS SSCs. The mean annual probability of exceedance (MAPE) is termed the hazard level in this document and in seismic design practice. Thus, a lower hazard level indicates a lower MAPE or higher amplitudes of ground motion. For example, a hazard level of 10^{-3}/yr would be associated with lower amplitude of ground motion than a hazard level of 10^{-4}/yr. Design codes, standards, and acceptance criteria were also specified in the initial issue of STR#2 to be those associated with applicable parts of NUREG-0800 (NRC 1987, Chapter 3). STR#2 also included a strategy for the mitigation of fault displacement hazards, which included criteria for fault avoidance and, in those cases where Type I faults (defined in McConnell et al. 1992, Section 3.1.3) cannot be avoided, the report described criteria and procedures for fault displacement design. The initial issue and first revision of the second topical report was reviewed by the NRC staff and, after comment resolution, the reviewers concluded that there were no further questions related to it, pending review of the three proposed topical reports.

The seismic design methodology in STR#2 (YMP 1997b) was subsequently updated in *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (DOE 2004) to be consistent with 10 CFR Part 63, *Yucca Mountain Review Plan*, *Final Report* (YMRP) (NRC 2003a), and recent regulatory actions regarding seismic design for nuclear facilities (Section 2.3). This document defines a risk-informed approach to establishing seismic DBGM levels, reaffirms the commitment to use NUREG-0800 (NRC 1987, Chapter 3) codes and standards, and defines the approach to demonstrating compliance using a seismic margin assessment (SMA). The document was submitted to NRC for review and a letter from the NRC on January 24, 2006, (Kokajko 2006) provided the staff review of the document.

Kokajko (2006) responded to the proposed methodology by drawing the following conclusions: (1) the seismic design bases and design codes and standards appear to be consistent with the regulatory requirements of 10 CFR 63.112(f)(2), (2) the SMA approach is useful but is not a substitute for demonstrating compliance with the performance objectives of 10 CFR 63.111(b)(2), and (3) additional supporting analyses are required to demonstrate compliance. The additional supporting analyses described include assessing the probabilities of seismic event sequences through the convolution of hazard curves and fragility curves, and evaluating whether the probability of unacceptable seismic performance of individual ITS SSCs is less than 1 in 10,000 over the preclosure period, as defined in 10 CFR 63.111(b)(2). Kokajko notes that if the probability of unacceptable seismic performance of individual ITS SSCs is greater than or equal to 1 in 10,000 over the preclosure period, the DOE may demonstrate compliance with 10 CFR 63.111(b)(2) by (1) showing the dose consequence is less than 5 rem, (2) showing that the probability of the complete event sequence is less than 1 in 10,000 over the preclosure period, or (3) modifying the design.

This revision to STR#2 provides a methodology that is responsive to Kokajko (2006). In particular, the methodology now includes probabilistic seismic analyses to support a compliance demonstration with the preclosure performance objectives of 10 CFR 63.111(a), (b), and (c) . The probabilistic seismic analyses (described in Section 4) incorporate elements of probabilistic risk analysis technology to demonstrate compliance for risk-significant SSCs.

A third seismic topical report (STR#3) was originally planned to describe the implementation of the methodologies described in STR#1 and STR#2 to develop seismic inputs for preclosure design and for postclosure performance assessment. The DOE has provided the information originally intended for inclusion in STR#3 in another document, *Technical Basis Document No. 14: Low Probability Seismic Events* (BSC 2004a), and supported by technical data in two additional reports, *Development of Earthquake Ground Motion Input for Preclosure Seismic Design and Postclosure Performance Assessment of a Geologic Repository at Yucca Mountain, NV* (BSC 2004b), and *Characterize Framework for Seismicity and Structural Deformation at Yucca Mountain, Nevada* (BSC 2004c). These documents cover the information originally intended to be included in STR#3. They describe the results of the PSHA for Yucca Mountain, the methodology to develop seismic design inputs based on the PSHA results, examples of implementing the methodology, and a brief overview of how seismic inputs will be developed and used in postclosure performance analyses. Final seismic inputs for LA design will be presented in the LA and supporting calculations.

1.2 PURPOSE AND SCOPE

The purpose of this report is to describe the seismic design and compliance demonstration methodology that the DOE will use. The approach integrates preclosure seismic design methodology, seismic margin assessment, and probabilistic seismic analyses to demonstrate compliance with the preclosure performance objectives contained in 10 CFR 63.111(a), (b), and (c), consistent with the risk-informed performance-based framework of the regulation. This report describes a design methodology intended to guide future design activities.

10 CFR Part 63 and the *Yucca Mountain Review Plan, Final Report* (YMRP) (NRC 2003a, Section 2.1.1, p. 2.1-2) do not provide specific design criteria, nor do they provide guidance on how to demonstrate compliance with the safety standard. Rather, the regulation allows the DOE to define an appropriate approach. As stated in the YMRP (NRC 2003a, p. 2.1-2):

> No prescriptive design criteria are imposed in the Yucca Mountain Review Plan because 10 CFR Part 63 allows the U.S. Department of Energy to develop the design criteria and demonstrate their appropriateness. Thus, the U.S. Department of Energy has flexibility to use any codes, standards, and methodologies it demonstrates to be applicable and appropriate. This flexibility is necessary when implementing a risk-informed, performance-based regulation.

The NRC has issued a final version of the document, *Interim Staff Guidance - HLWRS-ISG-01. Review Methodology for Seismically Initiated Event Sequences* (NRC 2006), to supplement the YMRP (NRC 2003a). The interim staff guidance "describes one method that staff may use to review the seismic performance of SSCs ITS and frequency of occurrence of seismic event sequences, as required by the analysis described in 10 CFR 63.112 to demonstrate compliance with the performance objectives in 10 CFR 63.111(b)(2)."

As documented in this report, the DOE has developed a comprehensive methodology that incorporates seismic design bases, seismic margin demonstration, and compliance demonstration. Seismic design bases, expressed as DBGM levels, are risk-informed and tied to the risk significance of ITS SSCs. The DBGM levels are consistent with regulatory precedent and the levels used for nuclear facilities having similar risk significance. The proposed seismic design criteria, codes, and standards have been demonstrated through extensive experience from nuclear power plants to result in significant seismic margin. Seismic margins will be quantified using approaches that have regulatory precedent in seismic safety demonstrations for nuclear power plants. Compliance with the preclosure performance objectives of 10 CFR 63.111 will be demonstrated based on probabilistic seismic analyses. Consistent with a risk-informed approach, the DOE preclosure seismic design methodology is integrated with PCSA. Both design analyses and safety analyses are used to demonstrate compliance with the preclosure performance objectives in 10 CFR 63.111(a), (b), and (c).

1.2.1 Relation to Preclosure Safety Analysis

According to 10 CFR 63.2, *Preclosure safety analysis* means a systematic examination of the site; the design; and the potential hazards, initiating events and event sequences and their consequences (e.g., radiological exposures to workers and the public). The analysis identifies structures, systems, and components important to safety.

Evaluations of preclosure safety are made with respect to a reference design for the geologic repository operations area. Therefore, SSCs of the preclosure design are evaluated in the PCSA to identify those that are ITS, in accordance with the definition of ITS given in 10 CFR 63.2 . The seismic design methodology in this topical report uses a risk-informed methodology for establishing DBGM levels for those SSCs that have been determined to be ITS and that are involved in seismically-initiated event sequences. In addition to DBGM levels, this report provides the seismic design codes, standards, and acceptance criteria in NUREG-0800 (NRC 1987, Chapter 3) that will be used in design. Although not a demonstration of compliance, a seismic margin assessment will be performed to show that the major structures have adequate seismic margin, as defined in Section 3.3 of this document. In order to demonstrate that the seismic design bases comply with the preclosure performance objectives of 10 CFR 63.111, probabilistic seismic analyses will be conducted. These probabilistic analyses (Section 4) involve the probabilistic consideration of earthquake ground motions, seismic fragility or capacity of ITS SSCs, and seismically-initiated event sequences. These analyses will be part of the PCSA. As the design progresses, probabilistic seismic analysis evaluates event sequence probabilities and doses in order to compare with the regulatory requirements. Modifications are made to the design, as needed to assure compliance. This interaction between engineering design and PCSA is consistent with a performance-based risk-informed philosophy.

1.2.2 Relation to Preclosure Repository Design and Postclosure Performance Assessment

The result of exercising the seismic design methodology in this topical report will be DBGMs at appropriate hazard levels for the preclosure seismic design of ITS SSCs. The DBGMs are expressed as ground motion response spectra for appropriate mean annual probabilities of exceedance. The actual response spectrum for ground motions associated with a particular DBGM level depends on the specific location where it is applied and is developed as part of the ground motion inputs (BSC 2004b). For example, for the same annual probability of exceedance, the ground motions at the surface will differ from those at depth in the emplacement drifts. Depending on the location and configuration of a particular SSC, additional location-specific evaluations may be required (e.g., in-structure floor response spectra may be needed for design of an SSC within a building). After assignment to a particular DBGM hazard level and appropriate modification of the motions to make them location-specific, the ground motions will be incorporated into Project documents. Likewise, applicable elements of *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* (NRC 1987, Chapter 3) will become part of design requirements (Section 3.2). As discussed in Section 5, the fundamental approach to mitigating the effects of fault displacement will be avoidance of Quaternary faults and fault displacement hazard avoidance is achieved when the amplitude of displacement is low enough that an explicit fault displacement design is not necessary. If fault displacement hazard avoidance is not possible, design basis fault displacements will be incorporated into the design.

The methods described for seismic design of ITS SSCs in this topical report are specifically applicable to preclosure seismic design and to demonstrating compliance with preclosure performance objectives of 10 CFR Part 63.111. There are no explicit seismic design requirements for postclosure, but the effects of seismic hazards (vibratory ground motion and fault displacement) on postclosure performance assessment are being evaluated in the total system performance assessment for a license application.

1.2.3 Assumptions and Limitations

This document presents a methodology and design criteria only, without models or analyses. Thus, there are no preestablished assumptions or limitations to the methodology given in this document.

1.2.4 Quality Assurance

This document is subject to the Office of Civilian Radioactive Waste Management *Quality Assurance Requirements and Description* (DOE 2006) as implemented by the *Quality Management Directive* (BSC 2007) and was prepared according to PA-PRO-0313, *Technical Reports*. This report describes a design methodology intended to guide future design activities but is not itself a design document. As such, there are no applicable design inputs, interfaces, analyses, test equipment, SSCs, or specified controls. This document provides a design approach description that will not be used for procurement. This document was developed per the Technical Work Plan, *Seismic Studies* (BSC 2006, Section 1.2).

1.3 REPORT ORGANIZATION

Following the Section 1 introduction, Section 2 provides a summary of the regulatory framework for the seismic design methodology outlined in this document. The framework includes the applicable regulations specific to Yucca Mountain, NRC regulatory precedents for other nuclear facilities, and seismic design practice for nuclear facilities not regulated by the NRC. Section 3 provides the seismic DBGM levels to be invoked for ITS SSCs and outlines the design codes, standards, and acceptance criteria in NUREG-0800 (NRC 1987, Chapter 3) that will be followed. The section also summarizes the analyses that will be conducted to ensure the seismic design criteria will lead to adequate seismic margins. Section 4 provides the probabilistic seismic analyses that will be conducted to demonstrate compliance with the preclosure performance objectives. Approaches to mitigate fault displacement hazards are given in Section 5. Section 6 summarizes the conclusions of this report and the references are provided in Section 7.

Appendix A describes the details for development of the high confidence capacity and permissible drift limits of low-rise concrete shear walls. Appendix B provides an explanation for abbreviations and acronyms.

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2. REGULATORY FRAMEWORK FOR SEISMIC DESIGN AND PERFORMANCE DEMONSTRATION METHODOLOGIES

This section describes NRC regulations, regulatory guidance, and regulatory actions regarding seismic design methodologies for Yucca Mountain and NRC-regulated nuclear facilities. DOE criteria and approaches to establishing seismic design levels for nuclear facilities are also described.

2.1 10 CFR PART 63 *DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN A PROPOSED GEOLOGIC REPOSITORY AT YUCCA MOUNTAIN, NEVADA*

10 CFR 63.41(c) specifies that the issuance of a license to receive and possess HLW requires a demonstration that the geologic repository operations area will not constitute an unreasonable risk to the health and safety of the public. The measure of acceptable risk is expressed in terms of allowable consequences for particular categories of event sequences. Allowable consequences are given as performance objectives (i.e., dose limits and numerical guides for design objectives) in 10 CFR 63.111 for Category 1 and Category 2 event sequences as defined in 10 CFR 63.2, which defines event sequences as:

> …a series of actions and/or occurrences within the natural and engineered components of a geologic repository operations area that could potentially lead to exposure of individuals to radiation. An event sequence includes one or more initiating events and associated combinations of repository system component failures, including those produced by the action or inaction of operating personnel.

The PCSA will identify event sequences. An event sequence is identified as beginning with an initiating event (from an identified hazard) that is followed by one or more events that must occur to result in a release of radioactivity, criticality, or an abnormal exposure to a worker. Event sequence categorization is based on the mean frequency of the entire sequence of events and not just the frequency of the initiating event. Using the definitions from 10 CFR 63.2, event sequence categories are quantified as:

- **Category 1**–Event sequences expected to occur one or more times before permanent closure
- **Category 2**–Event sequences with at least one chance in 10,000 of occurring before permanent closure

Categorizing event sequences is important because it establishes the portion of the preclosure performance objectives (10 CFR 63.111) that must be met for an event sequence. As will be discussed in Section 3.1, the application of the seismic design bases will preclude the occurrence of any seismically initiated event sequence having a mean annual probability of 10^{-3} or greater and, therefore, precludes the occurrence of a Category 1 seismically initiated event sequence.

2.1.1 Preclosure Performance Objectives

Category 1 Event Sequences

A limit of 0.15 mSv/yr at the site boundary and beyond is defined for Category 1 event sequences by 10 CFR 63.111(b)(1) as:

> (1) The geologic repository operations area must be designed so that, taking into consideration Category 1 event sequences and until permanent closure has been completed, the aggregate radiation exposures and the aggregate radiation levels in both restricted and unrestricted areas, and the aggregate releases of radioactive materials to unrestricted areas, will be maintained within the limits specified in paragraph (a) of this section.

And by 10 CFR 63.111(a) as:

> *(a) Protection against radiation exposures and releases of radioactive material.*
>
> (1) The geologic repository operations area must meet the requirements of part 20 of this chapter.
>
> (2) During normal operations, and for Category 1 event sequences, the annual TEDE (hereafter referred to as "dose") to any real member of the public located beyond the boundary of the site may not exceed the preclosure standard specified at § 63.204. [Note: TEDE = total effective dose equivalent]

And by 10 CFR 63.204 as:

> DOE must ensure that no member of the public in the general environment receives more than an annual dose of 0.15 mSv (15 mrem) from the combination of:
>
> (a) Management and storage (as defined in 40 CFR 191.2) of radioactive material that ...
>
> (b) Storage (as defined in § 63.202) of radioactive material inside the Yucca Mountain repository.

The dose limits given in 10 CFR Part 20, Subparts C and D, must also be maintained for Category 1 event sequences.

Category 2 Event Sequences

The limits for Category 2 event sequences are defined in 10 CFR 63.111(b)(2):

> (2) The geologic repository operations area must be designed so that, taking into consideration any single Category 2 event sequence and until permanent closure has been completed, no individual located on, or beyond, any point on the boundary of the site will receive, as a result of the single Category 2 event sequence, the more limiting of a TEDE of 0.05 Sv (5 rem), or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose

> equivalent may not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin may not exceed 0.5 Sv (50 rem).

2.1.2 Preclosure Safety Analysis

According to 10 CFR 63.21(c)(5), the safety analysis report in the LA must include:

> A preclosure safety analysis of the geologic repository operations area, for the period before permanent closure, to ensure compliance with §63.111(a), as required by §63.111(c)…

10 CFR 63.102(f) further describes the PCSA, including consideration of initiating events for event sequences, and emphasizes that initiating events are considered for inclusion only if they are reasonable:

> (f) *Preclosure safety analysis.* Section 63.111 includes performance objectives for the geologic repository operations area for the period before permanent closure and decontamination or permanent closure, decontamination, and dismantlement of surface facilities. 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.

In specifying the requirements for the PCSA, 10 CFR 63.112(b) makes it clear that the potential initiating events for consideration in the event sequences are naturally occurring events such as earthquake-related effects:

> (b) An identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area, including a comprehensive identification of potential event sequences; …

Finally, a key function of the PCSA is to identify those SSCs that are ITS. Per 10 CFR 63.112(e):

> (e) An analysis of the performance of the structures, systems, and components to identify those that are important to safety. This analysis identifies and describes the controls that are relied on to limit or prevent potential event sequences or mitigate their consequences. This analysis also identifies measures taken to ensure the availability of safety systems.

Those SSCs that have been determined to be ITS and credited with preventing or mitigating the consequences of a seismically initiated event sequence will be those subject to the seismic design methodology described in this report.

2.1.3 Design Bases and Design Criteria

Per 10 CFR 63.21(c)(3), the safety analysis report in the license application must include:

> (3) A description and discussion of the design of the various components of the geologic repository operations area and the engineered barrier system including:
>
> > (i) Dimensions, material properties, specifications, analytical and design methods used along with any applicable codes and standards;
> >
> > (ii) The design criteria used and their relationships to the preclosure and postclosure performance objectives specified at § 63.111(b), § 63.113(b), and § 63.113(c); and
> >
> > (iii) The design bases and their relation to the design criteria.

For clarification, 10 CFR 63.2 defines design bases as:

> *Design bases* means that information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be constraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or requirements derived from analysis (based on calculation or experiments) of the effects of a postulated event under which a structure, system, or component must meet its functional goals. The values for controlling parameters for external events include:
>
> > (1) Estimates of severe natural events to be used for deriving design bases that will be based on consideration of historical data on the associated parameters, physical data, or analysis of upper limits of the physical processes involved…

Therefore, the PCSA identifies ITS SSCs by a systematic evaluation of event sequences, including those with seismic-initiating events, and identifies design bases needed to ensure compliance with preclosure performance objectives.

10 CFR Part 63 does not provide specific guidance for an approach to prepare seismic analyses and design bases, nor does it provide guidance on how to demonstrate compliance with the safety standard. Rather than specifying design criteria or a methodology for analyses, the regulation allows the DOE to define an appropriate approach.

10 CFR 63.112(f) [specifies that the PCSA include:

> (f) A description and discussion of the design, both surface and subsurface, of the geologic repository operations area, including-
>
> > (1) The relationship between design criteria and the requirements specified at §63.111(a) and (b); and

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

This topical report directly responds to the requirement to provide seismic design bases and design criteria for those SSCs determined to be ITS. The implementation of the seismic design methodology and criteria given in the report will be documented in the license application.

2.2 YUCCA MOUNTAIN REVIEW PLAN

The YMRP (NRC 2003a) documents the NRC staff's expectations and emphasis during review of the license application, and, as such, provides a context for the development of a preclosure seismic design methodology and criteria that will meet the staff's expectations.

The YMRP (NRC 2003a, Section 2.1.1, p. 2.1-2) states that the development of specific design criteria is left to the DOE:

> No prescriptive design criteria are imposed in the Yucca Mountain Review Plan, because 10 CFR Part 63 allows the U.S. Department of Energy to develop the design criteria and demonstrate their appropriateness. Thus, the U.S. Department of Energy has flexibility to use any codes, standards, and methodologies it demonstrates to be applicable and appropriate. This flexibility is necessary when implementing a risk-informed, performance-based regulation. The risk-informed, performance-based review process in the Yucca Mountain Review Plan focuses on determining compliance with performance objectives as demonstrated by the U.S. Department of Energy preclosure safety analysis.

It is in this context that the seismic design criteria and methodology described in this document is presented for review and acceptance by the NRC.

In the process of identifying hazards and initiating events for the PCSA, the YMRP (NRC 2003a, Section 2.1.1.3.2, Review Method 1, p. 2.1-20) requests that the staff:

> Confirm that methods used to identify hazards and initiating events are consistent with Agency guidance or standard industry practices…If Agency guidance or standard industry practices are not used by the U.S. Department of Energy, evaluate whether the U.S. Department of Energy basis and justification for choosing a particular hazard and initiating event identification method are defensible.

Further, the staff is directed to verify that appropriate site-specific data have been used to identify naturally occurring hazards and initiating events, including seismicity and faulting (NRC 2003a, Section 2.1.1.3.2, Review Method 2, p. 2.1-20).

The YMRP directs the staff to "verify that design criteria and bases have been identified for structures, systems, and components important to safety" (NRC 2003a, Section 2.1.1.7.2.1, Review Method 1, p. 2.1-52). The adequacy of design bases and design criteria is to be judged based on the PCSA and, in particular, for seismic design consistency with NRC guidance (NRC 2003a, Section 2.1.1.7.2.1, Review Method 1, p. 2.1-53):

> Verify that design criteria adequately consider preclosure safety analysis results. Verify that structures, systems, and components important to safety will continue to prevent consequences, such as unacceptable releases of radioactive material, unacceptable radiation doses for workers or the public, and loss of removal capability.
>
> Confirm that structural design criteria and bases for structures, systems, and components important to safety are consistent with relevant U.S. Nuclear Regulatory guidance for tornado protection, seismic design, explosion protection, and flood protection.

Regarding fault displacement hazard, the design of subsurface operating systems can only be found adequate if "emplacement drifts are located away from major faults, consistent with the seismic design" (NRC 2003a, Section 2.1.1.7.3.3[ii], Acceptance Criterion 2, p. 2.1-70). In addition, the YMRP (NRC 2003a, Section 2.1.1.7.3.3[ii], Acceptance Criterion 4, p. 2.1-71) indicates that it should be shown that "the dynamic loads used in design analyses are consistent with seismic-design ground-motion parameters, consider faulting effects, and are consistent with accepted methodologies for assessing faulting hazards."

The YMRP (NRC 2003a) indicates that the staff's review of the PCSA will be risk-informed. Likewise, it is expected that the DOE PCSA will focus on ITS SSCs. Further, it is anticipated that ITS SSCs may be further distinguished by their relative risk significance (NRC 2003a, Section 2.1.1, p. 2.1-1 to 2.1-2):

> The structures, systems, and components important to safety may also be further categorized, based on relative safety significance, using risk information from the preclosure safety analysis. This distinction may be used to focus on the level of design details to be provided in the license application and the application of quality assurance controls.…
>
> The staff review is focused on items that the preclosure safety analysis has determined to be important to safety. The rigor of review for the design items on the Q-List, and the level of attention to detail, depend on relative safety significance.

The seismic design methodology described in this report is risk-informed. It will utilize the risk information developed in the PCSA to, first, identify the ITS SSCs and, second, to further define their risk significance such that the level of seismic DBGMs and level of fault displacement hazard are appropriate for the relative risk significance of ITS SSCs. As discussed in Section 3.1, the selection of the hazard levels associated with DBGM categories for ITS SSCs is based on a comparison of the risk significance of the Yucca Mountain preclosure facilities with those of other nuclear facilities. This approach ensures consistency with regulatory precedent and implements the risk-informed strategy called for in the YMRP (NRC 2003a).

2.2.1 Interim Staff Guidance - Review Methodology for Seismically Initiated Event Sequences

The NRC issued interim staff guidance for seismically initiated event sequences (NRC 2006) to supplement the YMRP (NRC 2003a). The interim staff guidance document "provides an example methodology to review seismically initiated event sequences, in the context of the preclosure safety analysis, for compliance with performance objectives in 10 CFR 63.111(b)(2)" (NRC 2006, p. 1). The suggested methodology is summarized by the following (NRC 2006, p. 1 to 2):

> The methodology considers the likelihood of seismic initiating events at the site, and the structural fragility of structures, systems, and components (SSCs) important to safety (ITS), to estimate probability of failure of SSCs ITS and frequency of occurrence of event sequences. This guidance was developed to take advantage of improvements in probabilistic seismic hazard analyses and performance-based safety assessments, thus differing from the design based and deterministic hazard criteria previously used for licensing of nuclear facilities, especially nuclear power plants.
>
> This ISG describes one method that staff may use to review the seismic performance of SSCs ITS and frequency of occurrence of seismic event sequences, as required by the analysis described in 10 CFR 63.112 to demonstrate compliance with the performance objectives in 10 CFR 63.111(b)(2). This methodology to evaluate seismic performance of an SSC ITS is similar to the one outlined in ASCE/SEI 43-05 (Ref. 2). NRC has accepted this methodology to support licensing of the mixed-oxide fuel fabrication facility at the Savannah River Site in South Carolina (Section 5.1.6.1 of Ref. 4). Application of the methodology described in ASCE 43-05 (Ref. 2) and the scope of seismic design and analysis for the GROA must be consistent with the Part 63 preclosure safety analysis requirements. The U.S. Department of Energy (DOE) may, however, use alternative methods to demonstrate compliance with the Part 63 preclosure safety analysis requirements for analysis of event sequences.
>
> [Note: For Ref. 1 see ASCE/SEI 43-05, for Ref. 4 see Duke Cogema Stone & Webster (2005).
> GROA = geologic repository operations area; ISG = interim staff guidance.]

The interim staff guidance document provides revisions to the YMRP in Section 2.1.1.4.2, "Review Methods" and Section 2.1.1.4.3, "Acceptance Criteria." Examples for exercising the methodology are given in appendices to the document, including the methodology for computing ITS SSCs probability of failure during a seismic event (NRC 2006, Appendix A), and the methodology for evaluating complete event sequences (NRC 2006, Appendix B). The methodology for probabilistic seismic analyses described in Section 4 of this report is considered consistent with the acceptable methodologies given in the interim staff guidance document.

2.3 SEISMIC DESIGN OF NUCLEAR POWER PLANTS AND OTHER NUCLEAR FACILITIES

An important regulatory context for the preclosure seismic design methodology for Yucca Mountain are regulations and regulatory guidance that are relied upon by the NRC and other agencies for determining the seismic safety at other nuclear facilities. In addition, recent NRC rulemakings and Commission statements during hearings provide additional insights into acceptable approaches to establishing seismic design methodologies.

2.3.1 Other NRC-Regulated Facilities

Current regulations require, and regulatory guidance provides for, PSHA as the fundamental means of characterizing the seismic environment at a site and the determination of appropriate seismic DBGM hazard levels based on a consideration of the risk significance of ITS SSCs. Deterministic approaches to defining seismic design levels for nuclear power plants (10 CFR Part 100, Appendix A) have been replaced by approaches based on PSHA (10 CFR 100.23). To implement PSHA requirements, an annual probability of exceedance (synonymously referred to as a reference probability, design hazard level, or design earthquake) must be established for ITS SSCs. The establishment of these design basis hazard levels in NRC regulations is now risk-informed in that the DBGMs are higher (i.e., the annual probability hazard level is lower) for nuclear power plants than for other nuclear facilities such as independent spent nuclear fuel storage installations. These differences in the DBGM hazard levels are justified by the differences in the potential consequences or risk significance associated with seismically induced failure of the facilities.

Regulatory Guide 1.165, *Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion*, Appendix B, for new nuclear power plants states that the DBGM should be associated with a <u>median</u> probability of 10^{-5} per year. Taking into consideration the probability distribution of seismic hazard curves at typical locations, a <u>median</u> annual probability of 10^{-5} is approximately equivalent to a <u>mean</u> annual probability of exceedance (MAPE) of 10^{-4}. This observation has been cited in recent findings related to Private Fuel Storage's independent spent nuclear fuel storage installation.

Specifically, it was stated in Finding F.103 of NRC (2003b, p. 333):

> …In Regulatory Guide 1.165, based on an analysis of the SSEs for existing NPPs, the Staff established the appropriate Reference Probability to determine the SSE at future NPP sites in connection with the use of a PSHA approach under 10 CFR 100.23; the Reference Probability was determined to be a 1×10^{-5} MAPE [see Note below] (approximately equivalent to a 100,000-year return period). As the Staff explained, this Reference Probability, which is defined in terms of the median probability of exceedance, corresponds to a MAPE of 1×10^{-4}. That is, the same design ground motion that has a median reference probability of 1×10^{-5}, has a MAPE of 1×10^{-4}.

> [Note: Correction, the Reference Probability of 1×10^{-5} applies to a *median* value, rather than a *mean* value.
> SSE = safe shutdown earthquake; NPP = nuclear power plant.]

For nuclear-related facilities other than nuclear power plants, the NRC and its staff have approved seismic DBGMs that are probabilistically based and that have mean annual probabilities of exceedance in the range of 4×10^{-4} to 5×10^{-4}:

- NRC approved Private Fuel Storage's independent spent nuclear fuel storage installation seismic design on May 22, 2003 (NRC 2003b, pp. 4 and 326). The design was based on a seismic design basis event with a return period of 2,000 years (equivalent to a MAPE of 5×10^{-4}), based on an approved exemption to 10 CFR 72.102(f)(1) (Parkyn 1999).

- The Mixed Oxide Fuel Fabrication Facility utilized a PSHA and a seismic hazard exceedance mean annual probability criterion that envelops the 5×10^{-4} event (Ihde 2001, p. 1.3.6-23) with reference to DOE Criteria for Performance Category 3 (PC-3) (DOE-STD-1020-94).

- Foster Wheeler Environmental Corporation utilized a PSHA and a MAPE of 4.0×10^{-4} (a 2,500-yr mean return period) for the design of the Idaho Spent Fuel Facility based on DOE Criteria for PC-3 (Idaho Spent Fuel Facility 2001, pp. 2.6-34 and -35 and 2.7-1).

NRC adopted a final rule on September 16, 2003 (68 FR 54159), that makes significant revisions to 10 CFR Part 72 (Travers 2003). The revised regulation requires a PSHA for determining the seismic DBGMs for a site (Travers 2003, cover letter):

> The final rule will make the 10 CFR Part 72 regulations compatible with the 1996 revision to Part 100 that addressed uncertainties in seismic hazard analysis. Specifically, the final rule changes will require a new specific-license applicant for a dry cask storage facility located in either the western U.S. or in areas of known seismic activity in the eastern U.S., and not co-located with an NPP, to address uncertainties in seismic hazard analysis by using appropriate analyses, such as a PSHA or other suitable sensitivity analyses, for determining the design earthquake ground motion (DE).

The NRC also adopted a risk-informed approach to identifying the DBGM level, taking into consideration the differences in risk between a nuclear power plant and an independent spent fuel storage installation (ISFSI) or monitored retrievable storage (MRS) facility (Travers 2003, cover letter):

> The staff also believes that the potential radiological consequences of a seismic event at an ISFSI or MRS storing spent fuel in dry casks or canisters are substantially less than the potential consequences of a similar event at an NPP. Therefore, the final rule will allow an ISFSI or MRS applicant to use a design earthquake level commensurate with the risk associated with an ISFSI or MRS, and thus the rule will be risk-informed and complies with the Commission's policies on probabilistic risk assessment and performance goals. The

> accompanying Regulatory Guide 3.73…recommends an acceptable design earthquake level. The staff's analysis and the basis for the recommendation is provided in the White Paper entitled, "Selection of the Design Earthquake Ground Motion Reference Probability"…

The NRC staff request to the Commission for approval of the final rule (Travers 2003, Attachment 4, the white paper referred to in the rulemaking above), provides an evaluation of the risk posed by an ISFSI or MRS facility and compares the potential earthquake-induced consequences with those that could occur at a nuclear power plant. It is concluded in that document that because the risks associated with an ISFSI or MRS are less than those posed by a nuclear power plant, the DBGMs (termed the design earthquake or DE) should, following a risk-informed regulatory policy, likewise be lower or less severe (Travers 2003, Attachment 4, p. 6):

> In a risk-informed, performance-based approach, the earthquake design level of the facility is selected based on the degree of risk associated with the facility.

The mean annual probability associated with the DBGMs is identified as 5×10^{-4} (2,000-yr return period) for an ISFSI or MRS, and this design level is compared with design levels for other facilities (Travers 2003, Attachment 4, Section 3.4, p. 12):

1. Based on the fact that the risk from an earthquake at a dry cask ISFSI or MRS facility is lower than at an NPP, the reference probability for such a facility should be higher than the reference probability of 1E-4 for an NPP. In other words, the design-mean-earthquake return period for such a facility should be less than 10,000 years.

2. The reference probability of 5E-4 (2,000-yr return period), for an ISFSI or MRS facility DE, is consistent with that used in DOE-STD-1020-2000, for similar-type facilities.

3. The *International Building Code 2000* (ICC 2000) requires the buildings, similar to a dry cask ISFSI or MRS facility, to be designed for earthquakes for a return period varying from 500 yrs to 1,300 yrs. Therefore, the recommended reference probability of 5E-4 (2,000-yr return period) provides more stringent seismic design criteria than International Building Code-2000 seismic design requirements.

2.3.2 DOE Nuclear Facilities

The DOE has implemented a risk-informed graded approach to seismic design in their *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities* (DOE-STD-1020-2002). SSCs ranging from nuclear reactors, nuclear facilities without reactors, essential buildings, and conventional buildings are categorized into four performance categories according to their risk significance. The DBGM probability levels for seismic design, as well as the performance goals, are tied to the performance categories. The PC-3 (DOE-STD-1020-2002, p. 2-4) indicates a design basis of 4×10^{-4} mean seismic hazard exceedance level (2,500-yr return period) for sites away from active tectonic plate boundaries (where the slopes of hazard curves may be different). Design forces are multiplied by a scale factor of 0.9 to bring the DBGM hazard levels to approximately 5×10^{-4} (2,000-yr return period).

2.3.3 ASCE Standard ASCE/SEI 43-05

The American Society of Civil Engineers/Structural Engineering Institute has published ASCE Standard *Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities* (ASCE/SEI 43-05, Sections 1.1 and 1.2). The stated intent of the standard is to provide a rational basis for the performance-based, risk-consistent seismic design of SSCs in nuclear facilities. DOE-STD-1020-94 and subsequent revisions initiated the concept of probabilistic design in that seismic performance categories for SSCs were established. The performance categories were each tied to a probabilistic target performance goal that represents a target mean annual frequency of unacceptable performance. ASCE/SEI 43-05 also provides criteria as a function of four limit states, the permissible deformation limit for each SSC established from functional considerations. Using a graded risk-informed approach, four seismic design categories are given for classifying SSCs based on their importance and failure consequences. Each seismic design category has a numerical target performance goal specified. Four limit states are defined as A, B, C, or D, (where A is just short of collapse and D is essentially elastic behavior). The standard (ASCE/SEI 43-05) specifies design criteria for load combinations including earthquake ground shaking (i.e., stress, displacement, and ductility limits) such that these limit states are not exceeded.

2.4 NUREG-0800, STANDARD REVIEW PLAN

NUREG-0800 (NRC 1987) provides guidance to the Office of Nuclear Reactor Regulation staff that is responsible for the review of applications to construct and operate nuclear power plants. NUREG-0800 applies only to nuclear power reactors and is not applicable to geologic repository systems. However, as discussed in Section 2.2, the YMRP (NRC 2003a) does not provide specific seismic design acceptance criteria. For this reason, the DOE has evaluated the sections of Chapter 3 of NUREG-0800 (NRC 1987, NRC 2007a,; NRC 2007b,; NRC 2007c,; NRC 2007d) that directly relate to seismic design methodology for potential applicability to geologic repository systems. Section 3.2 provides specific standard review plan sections and their applicability.

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3. DESIGN BASIS GROUND MOTIONS AND SEISMIC DESIGN

Seismic safety is achieved through a combination of two important design aspects: (1) the DBGM level, which establishes the amplitude of the ground motions that should be used for design, and (2) the conservatism in the design codes, standards, and acceptance criteria. The DOE has the flexibility to choose whatever seismic design bases and design procedures it believes will allow the NRC to find with reasonable assurance that the preclosure performance objectives of 10 CFR Part 63 have been met. The use of appropriate levels of DBGMs, adoption of applicable nuclear-power plant design methods (Section 3.2), and demonstration of adequate seismic margins beyond the DBGMs are part of the design methodology described below. This information is then used in conjunction with the probabilistic methodology described in Section 4 to demonstrate compliance with 10 CFR Part 63.

This section describes the DBGM levels that will be assigned for SSCs determined to be ITS and credited with preventing or mitigating the consequences of a seismically initiated event sequence. The assignment of DBGM levels is risk-informed such that SSCs determined in the PCSA to be more risk-significant will be subjected to more severe seismic design bases. Uncertainties in the ground motions associated with seismic design bases have been addressed and incorporated through the use of a probabilistic seismic hazard analysis (CRWMS M&O 1998). This section also describes the design codes, standards, and acceptance criteria in NUREG-800 (NRC 1987, Chapter 3) that will be used, which will ensure adequate levels of conservatism and design margin. Methods that will be used to demonstrate seismic margin are also discussed in this section. The levels of DBGM, design procedures to be used, and demonstration of seismic margin have been informed by precedents in the seismic design and evaluation of other nuclear facilities. For example, the annual probabilities of exceedance used for the DBGM levels are comparable to those employed for other facilities having similar risk significance (Section 3.1.1.1) and design codes, standards, and acceptance criteria are adopted from those used for nuclear power plants (Section 3.2). Section 3.3 presents a methodology for conducting a seismic margin assessment to demonstrate adequate seismic capacity against earthquake events for those specific ITS SSCs that are credited in an event sequence to demonstrate compliance. The seismic design basis ground motions presented in this document, in combination with the design criteria and margin demonstration, are consistent with a risk-informed methodology to achieve a proper design.

3.1 DESIGN BASIS GROUND MOTIONS

The seismic DBGM levels given in this document are associated with MAPEs. The amplitudes of ground motions associated with the DBGM levels are site-specific for the locations of repository components (i.e., surface and subsurface facilities) and are given in *Development of Earthquake Ground Motion Input for Preclosure Seismic Design and Postclosure Performance Assessment of a Geologic Repository at Yucca Mountain, NV* (BSC 2004b, Sections 6.3.1.2 and 6.3.1.3). The characterization of site-specific ground motions by specific parameters (e.g., accelerations, velocities, response spectra, time histories) is a function of the requirements for design implementation.

3.1.1 Design Basis Ground Motion Levels

Two DBGM levels will be used for the seismic design of ITS SSCs:

- DBGM-1 with a MAPE = 1×10^{-3} (1,000-year return period)
- DBGM-2 with a MAPE = 5×10^{-4} (2,000-year return period).

These DBGMs are defined based on risk significance of the YMP facilities. Two levels of DBGMs are applied in the risk-informed framework in accordance with prevention or mitigation of the two levels of performance objectives defined in 10 CFR 63.111(a) and (b), as discussed in Section 3.1.2.

To support the hazard levels associated with the two design basis levels, comparison can be made to the design bases of nuclear facilities having comparable risk significance (see Section 3.1.1.1). For example, Regulatory Guide 3.73, Section 3.4, specifies the DBGMs for independent spent fuel storage facilities as the ground motion associated with a MAPE of 5×10^{-4}. Likewise, DOE standard DOE-STD-1020-2002, Section 2, specifies the DBGMs for nonreactor nuclear facilities away from tectonic plate boundaries as the ground motion associated with a MAPE of 4×10^{-4} (2500-year return period). Design forces are multiplied by a scale factor of 0.9 to bring the earthquake design levels to approximately a MAPE of 5×10^{-4} (2000-year return period) (DOE-STD-1020-2002, Section 2). The DBGM-2 hazard level is comparable to levels for nuclear facilities having similar risk significance as the Yucca Mountain preclosure facilities, which is discussed further in Section 3.1.1.1.

3.1.1.1 Comparison of Risk Significance of Yucca Mountain Preclosure Facilities with Other Facilities

The development of two-tiered design basis levels and establishment of design bases based on risk significance is consistent with the intent of a risk-informed regulatory policy. Risk significance is defined as the consequences of failure and, in this case, the consequences associated with failure due to a seismic event. For example, the consequences of failure of a nuclear power plant would be more severe than failure of a nonnuclear facility. Accordingly, a risk-informed policy would indicate that the nuclear power plant should have more severe seismic design bases (i.e., larger DBGM levels). Comparisons made during rulemaking by the NRC support the conclusion that the risk significance of the YMP facilities is comparable to that of an ISFSI and is less than that of a nuclear power plant or nuclear fuel processing facility.

During the course of rulemaking and the implementation of its risk-informed regulatory policy in licensing actions over the past several years, the NRC has provided its views regarding the relative risk significance of the YMP facilities, ISFSI/MRS, and nuclear power plants. The following items are examples of these views.

(A) A repository is a relatively simple facility compared to a nuclear power plant (60 FR 15180 , pp. 15186 to 15187):

> Regardless of the type or nature of the initiating event, the Commission believes that, for several reasons, both the variety of credible event sequences and the

> resulting potential consequences to members of the public will be somewhat limited at repository facilities. First, in comparison with a nuclear power plant, an operating repository is a relatively simple facility in which the primary activities are in relation to waste receipt, handling, storage, and emplacement. A repository does not require the variety and complexity of systems necessary to support an operating nuclear power plant.

(B) The consequences of an accident are less at a repository than at a nuclear power plant (60 FR 15180, p. 15187):

> Further, the conditions are not present at a repository to generate a radioactive source term of a magnitude that, however unlikely, is potentially capable at a nuclear power plant (e.g., from a postulated loss of coolant event). As such, the estimated consequences resulting from limited source term generation at a repository would be correspondingly limited. This conclusion is consistent with the results of the aforementioned preliminary risk assessment by DOE of a conceptual repository design at Yucca Mountain, Nevada.

(C) Independent spent fuel storage installations and MRS installations have less risk significance than a nuclear power plant (SECY-01-0178; Travers 2001, p. 4):

> An ISFSI facility does not have the variety and complexity of active systems necessary to support an operating NPP. After the spent fuel is in place, an ISFSI facility is a static operation. During normal operations, the conditions required for the release and dispersal of significant quantities of radioactive materials are not present. Temperatures and pressures are relatively low during normal operations or even under design basis accident conditions; therefore, the likelihood of release and dispersal of radioactive materials is low primarily due to low heat generation rates of spent fuel with greater than the required one year of decay before storage in an ISFSI, combined with low inventory of volatile radioactive materials readily available for release to the environs. The long-lived and potentially biologically hazardous materials present in spent fuel are tightly bound up in the fuel materials and are not readily dispersible. The short-lived volatile nuclides, such as I-131, are no longer present in aged spent fuel (e.g., cooled at least one year). Furthermore, even if the short-lived nuclides were present during an event of a fuel assembly rupture, the canister surrounding the fuel assemblies would confine these nuclides. The radiological risk associated with an ISFSI facility is significantly less than the risk associated with an NPP, and therefore, the use of a lower design earthquake ground motion is appropriate.

(D) The hazards at a repository are similar to those at an ISFSI or MRS installation (64 FR 8640, p. 8644):

> This final dose limit, used in this regulation, is adapted from the dose limits specified in 10 CFR Part 72, for effluents and direct radiation during normal operations and anticipated operational occurrences, associated with a monitored retrievable storage installation (MRS). Like an MRS facility, the operations area

> at Yucca Mountain is expected to be a large industrial facility equipped to handle the loading, unloading, and decontamination of spent fuel and HLW shipping casks; the removal and packaging or repackaging of spent fuel assemblies and HLW canisters; and the sealing, handling, transport, stowage and periodic monitoring of canisters to contain the spent fuel and HLW during operations. Because the activities contemplated for the operations area prior to repository closure pose similar radiological hazards, during normal operations and anticipated operational occurrences, to those posed at an operating MRS, the Commission is proposing that the dose limits for the operations area be comparable to those applicable for the MRS, from planned discharges and from direct radiation during operations.

(E) Again, the hazards at a repository are similar to an independent spent fuel installation or MRS installation, stemming from the Private Fuel Storage (PFS) hearings (NRC 2005, pp. 6 to 8):

> The Commission's ruling compared the one-in-a-million threshold standard established for a GROA – a temporary storage area to be used in conjunction with a permanent repository for disposing of spent nuclear fuel – to the one-in-ten-million threshold standard established for a nuclear power reactor. The decision noted that in terms of both everyday operation and potential accident consequences, PFS's proposed ISFSI resembles a GROA more than a nuclear power reactor.[23] In addition, it pointed out that in previous rulemakings the NRC had announced its intent to "harmonize" regulations pertaining to ISFSIs and GROAs[24]…
>
> As the Commission held in 2001, in rulemakings prior to this adjudication it was made clear that GROAs and ISFSIs are similar facilities and should have the same design bases.[29] The Commission stated that there is "little basis" for using a reactor-like probability standard at an ISFSI (or a GROA); an accident at a reactor poses a greater risk than the accidental release of stored spent fuel because the contents of the reactor are under pressure that presents a "driving force behind dispersion" of radioactive materials[30] .
>
> [23] CLI-01-22, 54 NRC at 264-65. (see NRC 2001, pp. 11-14)
>
> [24] *See id.* at 264, citing 61 Fed. Reg. 64257, 64262 (Dec. 4, 1996). (see NRC 2001, pp. 11-12)
>
> [29] CLI-01-22, 54 NRC at 264. (see NRC 2001, pp 11-12)
>
> [30] Id. at 264-65. (see NRC 2001, pp. 11-14)

A reference earthquake that has a ground motion level associated with a MAPE of 5×10^{-4} (a return period of 2,000 years) is appropriate for the design of independent spent fuel installations or MRS installations (Travers 2003, Attachment 4, Section 3.4, p. 12):

1. Based on the fact that the risk from an earthquake at a dry cask ISFSI or MRS facility is lower than at an NPP, the reference probability for such a facility should be higher

than the reference probability of 1E-4 for an NPP. In other words, the design-mean-earthquake return period for such a facility should be less than 10,000 years.

2. The reference probability of 5E-4 (2,000-year return period), for an ISFSI or MRS facility DE, is consistent with that used in DOE-STD-1020, for similar-type facilities.

The YMP surface and subsurface nuclear facilities are designed with no high pressure or high temperature systems (DOE 2002, Sections 2.1.2.1 and 2.1.2.2), such as those that are common to nuclear power plants whose failure could lead to active energetic dispersal of radionuclides. The DOE concludes that the risk significance of the Yucca Mountain preclosure facilities is less than nuclear power plants and comparable to that of spent nuclear fuel storage facilities. Accordingly, the DOE has concluded that the use of DBGM-1 and DBGM-2 ground motion levels for the design of the surface and subsurface ITS SSCs is reasonable and appropriate.

3.1.2 Assignment of Design Basis Ground Motion Levels

The assignment of appropriate DBGM levels for specific SSCs is based on the risk significance of the associated event sequence. The assignment of the respective DBGM levels to ITS SSCs credited in the prevention or mitigation of a seismically initiated event sequence is tied to dose limits established in the performance requirements of 10 CFR 63.111, with reference to 10 CFR 63.204 and 10 CFR Part 20, for Category 1 and Category 2 event sequences. Table 3-1 summarizes the bases for DBGM assignments to ITS SSCs.

The assignments of DBGM-1 and DBGM-2 ground motion levels are based on a conservative estimate of the consequence of unmitigated release due to a seismically initiated event sequence, termed an unmitigated dose. Specifically, in assessing the potential dose consequence of the event sequence, no credit is taken for any active system or for aspects of confinement that would mitigate the release and thereby, mitigate the total dose or exposure to the public or workers. It should be noted that this conservative consequence analysis is used solely for purposes of DBGM assignment. Development of event sequences for the probabilistic seismic analyses is described in Section 4.6.1.

The specific assignment of DBGM levels for design is performed as part of the event sequence identification process and screening process described in more detail in Section 4.6.2. For SSCs credited as ITS in an event sequence (i.e., the SSCs whose failure to perform their intended safety function are the major contributors to risk in the seismically initiated event sequence), DBGM-1 is assigned to an SSC, as a minimum, if the seismic failure of the SSC, meaning a loss of its safety function, may result in a dose greater than the dose limits in Table 3-1.

Similarly, DBGM-2 is assigned to an SSC if the seismic failure of the SSC may result in a dose equal to or greater than 5 rem to the public at the boundary of the site or beyond. As described in Section 4.6.4, should it be shown that potential concurrent seismic failures of multiple SSCs initially assigned to DBGM-1 may result in a public dose of 5 rem TEDE or more, then one or more of the SSCs shall be reassigned to the DBGM-2 category until the potential dose of the multiple failure of remaining DBGM-1 SSCs is less than 5 rem TEDE.

There are special conditions in this approach. Specifically, assignments of DBGM-1 can be increased to DBGM-2, in a limited number of cases, to provide more in-depth defense of the repository.

Further, the assignment of DBGM levels will be extended as necessary to portions, parts, subparts or subsystems of an SSC when the response of such items could adversely affect the safety function performance of an ITS SSC in a seismically-induced event sequence. This type of sequence of events is typically termed a seismic interaction. The potential seismic interaction of a non-ITS SSC (source) with an ITS SSC (target)[2] shall require the assignment of the DBGM level of the target to the source, unless one of the following conditions can be demonstrated:

1. The interaction does not strike or significantly damage the ITS component, and therefore does not impair the performance of the safety function of the ITS SSC[3];

2. The event sequence involving the interaction has a probability of less than 1 chance in 10,000 over the preclosure period (without reliance on the non-ITS SSC to mitigate or prevent the sequence);[4] or

3. The consequence of the event sequence involving the interaction does not result in a dose in excess of the 10 CFR 63.111 performance standards.

If none of the above conditions are met, a preclosure safety requirement shall be applied, and the relevant portions, parts, subparts or subsystems of the source SSC(s) shall be required to be designed to the same seismic DBGM level as the target SSC. However, the source SSC may or may not be designated as ITS, as considered appropriate.

3.2 DESIGN CODES, STANDARDS, AND ACCEPTANCE CRITERIA IN NUREG-0800

NUREG-0800 (NRC 1987, Chapter 3), in general, ensures the quality and uniformity of NRC staff review and, in some cases, complements regulatory guides by providing a basis acceptable to the staff that may be used to implement requirements of NRC regulations. In particular, NUREG-0800 (NRC 1987, Chapter 3) identifies the regulations that are applicable to the seismic design of nuclear power reactors and identifies specific acceptance criteria, regulatory guides, and industry standards that provide information, recommendations, and guidance for compliance. The DOE considers that specific acceptance criteria and guidance provided by NUREG-0800 (NRC 1987, Chapter 3) are appropriate for use in preclosure seismic design.

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[2] This also applies to the potential interaction of an ITS SSC with a DBGM-1 assignment (source) with an ITS SSC with an assignment of DBGM-2 (target).

[3] Evaluations of the interaction shall include the dynamic loads and displacements produced by both SSCs, and for electrical or piping systems, up to the first anchor point beyond the interface.

[4] In demonstrating that a seismically-initiated event sequence involving an interaction has a probability of less than 1 chance in 10,000, the source SSC may be isolated from the target SSC by the use of barriers, the relocation of the non-ITS SSC, or by the introduction of constraints or supports.

With exceptions as noted in the following text, the DOE considers that sections of the standard review plan (Section 3.7.1, "Seismic Design Parameters", NRC 2007a; Section 3.7.2, "Seismic System Analysis", NRC 2007b; Section 3.7.3, "Seismic Subsystem Analysis", NRC 2007c; Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment", NRC 2007d) provide appropriate codes, standards, and acceptance criteria for the preclosure ground motions design of repository surface facilities that are ITS. The exceptions are as follows:

- Where differentiated in NUREG-0800 (NRC 1987), requirements for documentation to be provided in the preliminary safety analysis report to support an application for the construction permit are appropriate for the geologic repository system LA. Requirements for documentation to be included in the final safety analysis report to support an application for an operating license are not appropriate for the geologic repository system LA. This documentation is developed during procurement and construction and would be available for NRC inspection prior to the issue of a license to receive and possess waste.

- Requirements for the design of those specific SSCs that are present in a nuclear power reactor, but which would not be present in repository surface facilities, do not apply.

Table 3-1 Bases for Assigning DBGMs to ITS SSCs

Performance Objectives Applied to Seismic Preclosure Safety	Dose Receptor	Potential Consequences of Loss of SSC Safety Function Dose Limit (TEDE) [a]	DBGM Assigned to ITS SSCs
Category 1 Event Sequences 10 CFR 63.111(b)(1) 10 CFR Part 20	Repository Employee Receiving an Occupational Dose	>5 rem (0.05 Sv)	DBGM-1
	Repository Employee Not Receiving an Occupational Dose Or Member of the Public Onsite Or Nevada Test Site and Nellis Workers in an Unrestricted Area	>100 mrem (1.0 mSv) [b,c] or >2 mrem (0.02 mSv) in one hour [b,c] or >10 mrem (0.1 mSv) from air emissions [c,d,f]	DBGM-1
	Member of the Public Beyond the Site Boundary in the General Environment [d]	>15 mrem (0.15 mSv)	DBGM-1
Category 2 Event Sequences 10 CFR 63.111(b)(2)	Individual at or Beyond the Site Boundary [e]	≥ 5 rem (0.05 Sv)	DBGM-2
Criticality Condition 10 CFR 63.112(e)(6)	N/A	N/A	DBGM-2

NOTES:

[a] Dose limits are aggregate doses for Category 1 Event Sequences and are single-event sequence doses for Category 2 Event Sequences. Values are for the higher of the TEDE (a measure of body dose) or sum of the deep dose equivalent and the committed dose equivalent. Higher dose equivalents for the lens of the eye, skin, and extremities are not included in the table, but are subject to separate limits per 10 CFR 63.111(b)(2) and 10 CFR Part 20, Subparts C and D.

[b] Dose limits do not include occupational dose or doses received from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released under 10 CFR 35.75, or from voluntary participation in medical research programs.

[c] Dose limits are taken as equal to the maximum annual dose limit where an accident-related dose is not specified.

[d] General environment means everywhere outside the Yucca Mountain site, Nellis Air Force Range, and Nevada Test Site.

[e] At any point on the site boundary.

[f] “As low as reasonably achievable” (ALARA) goal per 10 CFR 20.1101.

DBGM = design basis ground motion; ITS = important to safety; N/A = Not Applicable; SSCs = structures, systems, and components; TEDE = total effective dose equivalent.

- In general, 10 CFR Part 100, Appendix A, requirements for development of DBGMs do not apply. In particular, requirements for the operating basis earthquake and safe shutdown earthquake ground motions do not apply. The DOE will develop DBGM-1 and DBGM-2 as discussed in Section 3.1.

- Development of seismic time histories for use in design analyses and consideration of the variation in soil properties in soil-structure interaction analyses will generally follow the guidance in NUREG/CR-6728, *Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines* (McGuire et al. 2001, Section 5) and ASCE/SEI 43-05, *Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities* (ASCE/SEI, 2005, Section 2).

- References to seismic Category-1 SSCs, per NUREG-0800 (NRC 1987 , Chapter 3) terminology, will be treated as references to ITS SSCs in accordance with the definition of this term in 10 CFR Part 63.

Acceptance criteria from other sections of the standard review plan of NUREG-0800 (e.g., Section 3.8.4, "Other Seismic Category 1 Structures," NRC 1987); Section 3.9.1, "Special Topics for Mechanical Components," NRC 1987); and Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," NRC 2007e), will be evaluated for applicability to the repository surface and subsurface facilities.

3.3 SEISMIC MARGIN ASSESSMENT FOR STRUCTURES

In addition to the design of ITS SSCs to a specific design basis (either DBGM-1 or DBGM-2), the DOE intends to demonstrate seismic margins for the major structures against earthquake ground motions that are considerably larger than the DBGMs. The approach used is similar to seismic margin assessments that have been performed for nuclear power plants. Although not part of the compliance demonstration, the purpose of the seismic margin assessment will be to demonstrate that the major structures have adequate seismic margin, where "adequate" is defined as having a high confidence of low probability of failure (HCLPF) capacity that exceeds the designated review level earthquake, termed a beyond design basis ground motion (BDBGM) event for Yucca Mountain facilities.

Seismic margin assessments have considerable precedent in seismic evaluations of SSCs for nuclear power plants and have been used to demonstrate the adequacy of seismic margins for the NRC-regulated power reactor facilities with levels of risks to workers and the public that are comparable to or higher than those at Yucca Mountain. NUREG-1407 (Chen et al. 1991) identifies SMA and probabilistic risk analyses as acceptable risk-informed approaches for evaluating seismic safety and identifying seismic vulnerabilities, as required by the NRC Individual Plant Examination of External Events (IPEEE) program. NUREG-1742 (NRC 2002 , Table 2.1) indicates that 43 of 71 plant sites used the SMA approach in their IPEEE evaluations. The approach is based on a comparison of a conservative estimate of the capacity of the facility to maintain safety functions with the demand imposed by review level earthquake ground motions that are greater than the DBGMs. HCLPF capacities are assessed following the implementation guidance for SMAs given in *A Methodology for Assessment of Nuclear Power*

Plant Seismic Margin (Revision 1) (EPRI 1991a), including the use of the Conservative-Deterministic-Failure-Margins (CDFM) approach (see Section 4.4 for additional discussion). The HCLPF capacity is defined as the ground motion level at which there is a mean conditional probability of failure of 0.01 or less. ASCE 4-98, Appendix A, provides a discussion of the applicability of the SMA approach to demonstrate seismic safety of plants designed using NUREG-0800 (NRC 1987, Chapter 3) codes and standards. As discussed in Section 4.4, the HCLPF capacity assessments will also be used to develop fragility curves for the major structures for probabilistic seismic analyses for the compliance demonstration.

In implementing the seismic margins approach for Yucca Mountain structures, a review level earthquake has been selected that is consistent with the characteristics of the site and the implementation of SMA for nuclear power plants. The review level earthquake, termed BDBGM for the Yucca Mountain facilities, is associated with a mean annual exceedance probability of 10^{-4}. Consistent with implementation of the SMA approach for nuclear power plants, the review level earthquake loading should be sufficiently larger than the design basis to challenge the seismic margins of the facility. The ratio between the review level earthquake and design basis can be assessed by comparing the review level earthquakes given in NUREG-1742 (NRC 2002, Table 2.1) with the safe shutdown earthquakes given in NUREG-1488 (Sobel 1994, Appendix C). Examination of SMA IPEEE evaluations for nuclear power plants shows that the average ratios between the review level earthquake ground motions and design bases ground motions (peak ground acceleration [PGA] or peak spectral acceleration [PSA]) are approximately 1.5 to 1.9, respectively. The ratio for the Yucca Mountain surface facilities is approximately 2 (for both PGA and PSA), which is comparable and conservative with respect to the nuclear power plants evaluated using the SMA approach.

The seismic margins assessment will ensure that the HCLPF capacity of the major structures will exceed the BDBGM event (review level earthquake). This will ensure that adequate seismic design margins will exist for these structures, such that they will maintain their defined functions credited in the PCSA. This information will provide additional support to the compliance demonstration, which is discussed in Section 4.

4. PROBABILISTIC SEISMIC ANALYSES FOR COMPLIANCE DEMONSTRATION

4.1 BACKGROUND AND FRAMEWORK

As discussed in Section 1.1, the previous revision to this topical report (DOE 2004) provided a methodology for demonstrating seismic safety margins and compliance with preclosure performance objectives in 10 CFR 63.111 using an SMA approach. The DOE took the position that 10 CFR 63.102(f) provides for the use of preclosure safety methodologies that have precedents adopted for nuclear facilities with comparable or higher risks to workers and the public. Accordingly, DOE expressed its belief in *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (DOE 2004) that SMA, which has been used to evaluate the seismic safety of nuclear power plants, will provide an appropriate basis for assessing the preclosure seismic safety of the repository facilities. In response, Kokajko (2006) provided the NRC staff views of the DOE position regarding the compliance demonstration methodology. In summary, the NRC staff concluded that the SMA approach is useful but does not provide a substitute for demonstrating compliance with 10 CFR 63.111(b)(2). To do so, the NRC staff called for additional supporting analyses that would demonstrate that the calculated probability of seismically initiated event sequences of individual ITS SSCs is less than 1 in 10,000 over the preclosure period, as defined in 10 CFR 63.111(b)(2). This section of the report provides the methodology for probabilistic seismic analyses that will provide a demonstration of compliance consistent with Kokajko. Likewise, the probabilistic seismic methodology is consistent with the acceptable methodologies identified in *Final Version - HLWRS-ISG-01, Review Methodology for Seismically Initiated Event Sequences* (NRC 2006).

The DOE understands that the conclusions by Kokajko (2006) are limited to seismically initiated event sequences and believes that elements of the SMA approach in addition to probabilistic seismic analyses will demonstrate compliance with the regulations. Therefore, the DOE has modified the compliance demonstration methodology to include probabilistic seismic analyses for risk-significant SSCs. As discussed later in this section, screening analysis will be used to focus the analyses on risk-significant structures and components. The analyses will evaluate the probability of seismic event sequences and compare them to the lower Category 1 probability threshold of 1 or more incidents over the preclosure period and the lower Category 2 probability threshold of 1 in 10,000 during the preclosure period. Dose analyses will be done for those event sequences above the lower Category 2 probability threshold in order to compare against the regulatory dose limits. During the seismic design and probabilistic analysis process, design changes will be made such that compliance is achieved by either dose or probability reduction. It is recognized that the probabilistic seismic analysis described in this section need not be a full probabilistic risk assessment because each event sequence within Category 2, as opposed to the probabilistic sum of event sequences, is evaluated against the regulatory limits. This is consistent with Kokajko and preclosure performance objectives in 10 CFR 63.111(b)(2) for Category 2 event sequences.

4.2 OVERVIEW OF APPROACH

The overall approach to the probabilistic seismic analysis is summarized in Figure 4-1 and will follow standard practice as documented in numerous seismic risk assessment references (e.g., NRC 1983; Chen et al. 1991, Section 3.1; EPRI 1994; Kennedy et al. 1980; IAEA 1993; and ANSI/ANS-58.21-2007). The three key stages of the analysis are the seismic hazard development, fragility evaluation, and event sequence analysis (often called systems analysis).

The stages of the performance evaluation, summarized briefly below, are discussed in more detail in subsequent sections:

Seismic Hazard Development — Site-specific seismic hazard curves are required to represent the annual probability of various amplitudes of ground motion at the location of the surface facilities. The mean ground motions at particular annual probabilities of exceedance, expressed as uniform hazard spectra (UHS), are calculated using the site response model (BSC 2004b) and mean-centered representations of the uncertainty and variability in the inputs to the site response model. The site response calculations are conducted for mean annual probabilities of exceedance below 10^{-6} to provide seismic hazard curves that can be used in the compliance demonstration. The methodology for the development of the seismic hazard curves is given in Section 4.3.

Seismic Fragility Evaluation of SSCs — Seismic fragility analysis determines the conditional probability of failure as a function of an appropriate ground motion parameter (e.g., PGA, spectral acceleration). Failure is defined as the inability of an SSC to perform or provide its intended safety function. Mean fragility curves will be developed for specific ITS SSCs based on an assessment of the HCLPF capacity and composite logarithmic standard deviation, β (Kennedy 2001). The CDFM approach will be applied when a HCLPF capacity serves as a basis of a fragility curve and it is expected that structures will be analyzed using this method (EPRI 1991a and ASCE 4-98, Appendix A). A second approach (EPRI 1994) will be applied for components and the median capacity will serve as a basis of a mean fragility curve, which will use a composite logarithmic standard deviation.

Event Sequence Analysis and Screening — This stage includes identification and quantification of seismically initiated event sequences for comparison to the Category 1 and 2 thresholds. End states of the event sequences will be potential dose. The specific dose calculations will be in concert with the regulatory dose limits for Category 1 or Category 2 event sequences, as appropriate. Event sequences demonstrated by quantification of mean values to be below the lower Category 2 threshold will be screened out from further study. Event sequences with estimated doses that are below the regulatory limits of 10 CFR 63.111(b)(2) will be considered to be in compliance.

The results of the performance evaluation are the following: the identification, quantification, and categorization of seismically-initiated event sequences and associated consequences, and the assignment of DBGM levels DBGM-1 or DBGM-2 to ITS SSCs that are included in event sequences.

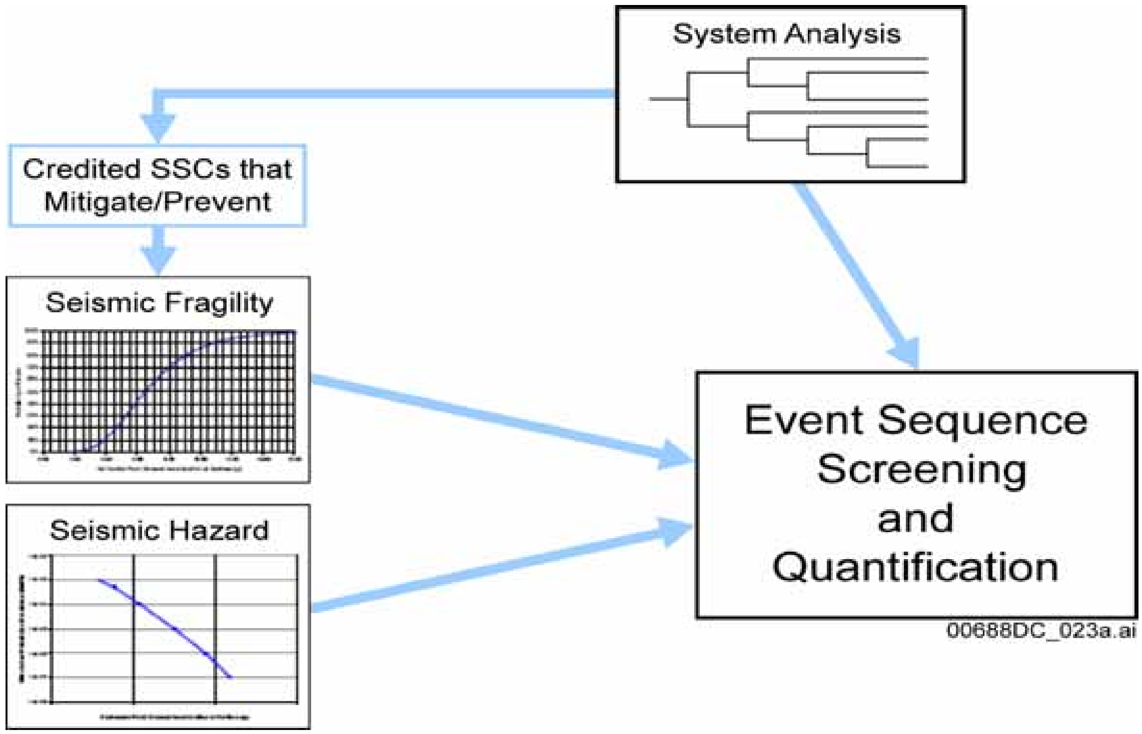

Figure 4-1. Overall Approach for Performance Evaluation

4.3 DEVELOPMENT OF MEAN HAZARD CURVES FOR PROBABILISTIC ANALYSES

For the probabilistic seismic analyses, the soil motions must be hazard consistent (i.e., the annual exceedance probability of the soil UHS should be the same as the rock UHS). In NUREG/CR-6728 (McGuire et al. 2001, Section 6.1), several site response approaches are recommended to produce soil motions consistent with the rock outcrop hazard. The approaches also incorporate the aleatory variabilities in the soil properties into the soil motions. To compute the ground motions for preclosure probabilistic seismic analyses, Approach 3 (McGuire et al. 2001, Section 6.1) will be implemented. Approach 3 is also described by Bazzurro and Cornell (2004) and NUREG/CR-6769 (McGuire et al. 2002, Section 6). This approach will also be used to develop ground motions for DBGM-1, DBGM-2, and BDBGM.

In this approach, the hazard at the soil surface is computed by integrating the site-specific hazard curve at the bedrock level with the probability distribution of the amplification function. The soil amplification is characterized by a suite of frequency-dependent amplification functions that can account for nonlinearity in soil response. Approach 3 involves approximations to the hazard integration using suites of transfer functions, which result in complete hazard curves at the ground surface (McGuire et al. 2002, Section 6) for specific ground motion parameters (e.g., spectral accelerations) and a range of frequencies.

The basis for Approach 3 is a modification of the standard PSHA integration:

$$P[A_S > z] = \iiint P\left[AF > \frac{z}{a} \mid m, r, a\right] f_{M,R|A}(m,r;a)\, f_A(a)\, dm\, dr\, da \qquad \text{(Eq. 4-1)}$$

where A_S is the random ground motion amplitude on soil at a certain natural frequency; *z* is a specific level of A_S; *m* is earthquake magnitude; *r* is distance; *a* is an amplitude level of the random rock ground motion, A, at the same frequency as A_S; $f_A(a)$ is derived from the rock hazard curve for this frequency (namely it is the absolute value of its derivative); and $f_{M,R|A}$ is the disaggregated hazard (i.e., the joint distribution of M and R, given that the rock ground motion amplitude is level a). AF is an amplification factor defined as:

$$AF = \frac{A_S}{a} \qquad \text{(Eq. 4-2)}$$

where AF is a random variable with a distribution that can be a function of *m*, *r*, and *a*. To accommodate epistemic uncertainties in site dynamic material properties, multiple suites of AF may be used and the resulting hazard curves combined with weights to properly reflect mean hazard and fractiles.

Soil response is controlled primarily by the level of rock motion and *m*, so Equation 4-1 can be approximated by:

$$P[A_S > z] = \iint P\left[AF > \frac{z}{a} \mid m, a\right] f_{M|A}(m;a)\, f_A(a)\, dm\, da \qquad \text{(Eq. 4-3)}$$

where *r* is dropped because it has an insignificant effect in most applications. To implement Equation 4-3, only the conditional magnitude distribution for relevant levels of *a* is needed. $f_{M|A}(m;a)$ can be represented (with successively less accuracy) by a continuous function, with three discrete values or with a single point, (e.g., $m^1(a)$, the mean magnitude given a). With the latter, Equation 4-3 can be simplified to:

$$P[A_S > z] = \int P\left[AF > \frac{z}{a} \mid a, m^1(a)\right] f_A(a)\, da \qquad \text{(Eq. 4-4)}$$

where, $f_{M|A}(m;a)$ has been replaced with $m^1(a)$ derived from deaggregation. With this equation, one can integrate over the rock acceleration, *a*, to calculate $P[A_S>z]$ for a range of soil ground motion amplitudes, *z*. However, an alternative solution is to use a closed-form approximation (McGuire et al. 2002, Appendix A; Bazzurro and Cornell 2004):

$$z_{rp} = a_{rp}\, \overline{AF_{rp}}\, \exp\left(\frac{1}{2} k \left[\frac{\sigma^2}{{d_3}^2}\right]\right) \qquad \text{(Eq. 4-5)}$$

where z_{rp} is the soil ground motion amplitude *z* associated with return period *rp*; a_{rp} is the rock ground motion amplitude associated with return period *rp*; $\overline{AF}_{rp}$ is the mean amplification factor (logarithmic mean) for the rock motion with return period *rp*; *k* and d_3 are the (log) slopes of the rock hazard curve and AF, respectively; and σ is the log standard deviation of AF.

The mean hazard curves developed for preclosure seismic analyses will also incorporate information related to reasonable bounds to extreme ground motions at Yucca Mountain.

4.4 DEVELOPMENT OF FRAGILITIES FOR PROBABILISTIC ANALYSES

A fragility curve provides the probability of unacceptable seismic performance as a function of a ground motion parameter such as PGA or dominant spectral acceleration. For the ITS SSCs identified as being important components of event sequences for compliance demonstration, permissible limit states will be defined per ASCE/SEI 43-05, Table 1-4). Seismic fragilities will be developed as a function of the limit states and ground motions using the methods described below.

4.4.1 Establishing Limits on Permissible Damage

Before developing a seismic fragility estimate for an SSC, it will be necessary to specify what constitutes unacceptable damage for each specific SSC in the event sequence. The unacceptable damage states will be defined in terms of the limit states given in ASCE/SEI 43-05, Table 1-4, as shown in Table 4-1.

Table 4-1. Limit State Classification and Structural Damage Comparison

Limit State	Structural Deformation Limit	Amount of Damage
A	Large Permanent Distortion, Short of Collapse	Significant
B	Moderate Permanent Distortion	Generally Repairable
C	Limited Permanent Distortion	Minimal
D	Essentially Elastic Behavior	None

The appropriate limit state will be selected for each SSC in each event sequence to achieve the desired safety function.

4.4.2 Development of Fragility Curves for Structures

Mean fragility curves for structures will be developed using the 1% conditional probability of failure seismic capacity, $C_{1\%}$, and the composite logarithmic standard deviation, β (EPRI 1991; Kennedy 2001, Sections 5 and 6). Other methods, such as the fragility analysis methods outlined in *Methodology for Developing Seismic Fragilities* (EPRI 1994, Section 4), may be used on a case-by-case basis.

The 1% conditional probability of failure seismic capacity will be approximated by the deterministically computed CDFM methodology (EPRI 1991, pp. 2-45 to 2-56; ASCE/SEI 43-05, Section C1.3, for 1% conditional probability of failure). The capacity obtained from the CDFM method is called C_{CDFM}. Alternatively, the capacity evaluation methodology (DOE-STD-1020-2002, Section C.5; Kennedy and Short 1994, Section 4.2) can be used to determine the C_{CDFM}. Kennedy (2001, Sections 3 and 5) shows that the high-confidence-of-low-probability-of-failure (HCLPF) capacity computed by the CDFM method closely approximates the 1% conditional probability of failure seismic capacity, $C_{1\%}$, point on the mean seismic fragility curve:

$$C_{HCLPF} \approx C_{CDFM} \approx C_{1\%} \quad \text{(Eq. 4-6)}$$

such that these capacity definitions may be used interchangeably.

The mean fragility curve will be defined as lognormally distributed with a $C_{1\%}$ capacity and logarithmic standard deviation, β. β is a composite standard deviation that includes both randomness (aleatory uncertainty) and epistemic uncertainty. Utilizing $C_{1\%}$ from the deterministic computations, the median capacity is given by:

$$C_{50\%} = C_{1\%} \exp(2.326\beta) \quad \text{(Eq. 4-7)}$$

where 2.326 is the number of standard normal variants that the 1% point lies below the 50% point (Kennedy 2001, Section 2.1.2 and Table 3).

The fragility composite logarithmic standard deviation, β, will be estimated by judgment following guidance in ASCE/SEI 43-05, Section C2.2.1.2. For example, structures and major passive mechanical components mounted on the ground or at low elevations within structures, β typically ranges from 0.3 to 0.5. For active components mounted at high elevations in structures, β typically ranges from 0.4 to 0.6.

The annual probability of failure to perform a safety function, P_F, for any SSC is relatively insensitive to β. This point is illustrated by Kennedy (2001, Section 5.3 and Table 4) and EPRI (1994, Section 5). Over the range of β from 0.3 to 0.6, the computed seismic risk differs by a factor of approximately 2.6. The computed seismic risk at $\beta = 0.3$ is approximately 1.5 times that at $\beta = 0.4$, while at $\beta = 0.6$ the computed seismic risk is approximately 60% of that at $\beta = 0.4$. An estimate of β is sufficient to estimate the seismic risk, P_F, within a factor of 1.6. Therefore, the annual probability of failure can be computed with adequate precision using $C_{1\%}$ and an estimate of β.

In summary, the complete mean fragility curve is defined by the $C_{1\%}$ capacity deterministically computed using the CDFM methodology and estimates of β.

4.4.3 Determination of CDFM Capacity

The CDFM capacity of any SSC can be estimated from:

$$C_{CDFM} = F_S * F_\mu * BDBGM \quad \text{(Eq. 4-8)}$$

where:

BDBGM = beyond DBGM for which the SSC has been evaluated

F_S = computed strength margin factor

F_μ = nonlinear margin factor.

4.4.3.1 Strength Margin Factor

The strength margin factor, F_S, is given by:

$$F_S = \frac{F_C C_C - D_{NS}}{D_S} \qquad \text{(Eq. 4-9)}$$

where:

C_C = capacity computed using code capacity acceptance criteria (including code-specified strength reduction factors ϕ)

D_{NS} = expected concurrent nonseismic demand

D_S = seismic demand computed for the BDBGM input in accordance with the requirements of ASCE 4-98, Section 3.1.1.2.

F_C = capacity increase factor (based on EPRI (1991, Equation 2-6); Kennedy (2001, Appendix A)).

$$F_C = \frac{C_{98\%}}{C_C} \qquad \text{(Eq. 4-10)}$$

where $C_{98\%}$ is the estimated 98% exceedance probability capacity.

The estimate of $C_{98\%}$ capacity for the shear strength of low-rise concrete shear walls will be based on ASCE/SEI 43-05, Section 4.2.3. A number of examples for estimating $C_{98\%}$ for other SSCs is given in *A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)* (EPRI 1991, Appendices L and M) and this guidance will be followed. When data are inadequate to estimate $C_{98\%}$ or for the sake of simplicity, F_C can be conservatively taken as 1.0.

Section A-1.1 of Appendix A describes the details for developing the high confidence shear strength capacity for low-rise concrete shear walls.

4.4.3.2 Nonlinear Margin Factor

In the CDFM method (Kennedy 2001, Section A.2.4; EPRI 1991, Table 2-5), the nonlinear margin factor, F_μ, is estimated at the 95% exceedance probability. Generic estimates of the 95% exceedance probability F_μ for SSCs are given in ASCE/SEI 43-05, Tables 5-1 and 8-1, for Limit States A, B, and C (F_μ values are unity for Limit State D). The corresponding drift and rotation limits are given in Tables 5-2 and 5-3, respectively, of ASCE/SEI 43-05. The basis for the low-rise concrete shear wall drift limits is presented in Section A-1.2 of Appendix A.

As an example, the lateral drift per story of a low-rise concrete shear wall (height to length ratio less than 2.0) is limited to less than 0.4% of the story height for Limit State C per ASCE/SEI 43-05, Table 5-2. Thus, for a 10-ft story height, the lateral drift is limited to 0.48 inches. This limit provides high confidence that shear cracks in the wall will be small and that the ultimate strength of the wall will not be reduced by a few cycles of plus and minus distortion carried to this drift limit. The wall retains its full strength and serviceability. This 0.4% of the story height drift limit is identical to the drift limit specified in DOE-STD-1020-2002, Section 2.3, for low-rise concrete shear walls.

4.4.4 Development of Fragility Curves for Equipment and Components

Equipment and component fragility analysis will follow two approaches. For those equipment and components that have analogues in nuclear power plants and typically have large capacities (e.g., air handling units, large switchgear, large transformers, horizontal motors, air handling units), representative fragilities will be used within the event sequences (e.g., Budnitz et al. 1985, Appendices B and C; Cover et al. 1985) to evaluate performance (see Section 4.6). The median-centered fragility analysis method (Kennedy and Ravindra 1984); EPRI 1994) will be applied for all other ITS equipment and components credited in a seismic event sequence to demonstrate compliance.

4.4.4.1 Calculating Fragilities

Similar to the CDFM method, an equipment fragility analysis starts with a stress calculation analogous to that performed to demonstrate that the code loads produce stresses within the code allowables. In a fragility analysis, the goal is somewhat different. It is to determine the median capacity as a function of a ground motion variable such as spectral acceleration or peak ground acceleration, and the uncertainty in this capacity. In the fragility calculation, a reference ground motion corresponding to a BDBGM will be used.

The median capacity, $\bar{a}$, may be expressed as follows (EPRI 1994, p. 4-6):

$$\bar{a} = \overline{SF} a_{ref} \qquad \text{(Eq. 4-11)}$$

In Equation 4-11, $\overline{SF}$ = the median scale factor and a_{ref} = the reference earthquake motion variable (e.g., PGA or spectral acceleration).

Having found the median capacity, the fragility curve is developed by estimating a composite logarithmic standard deviation, β, which includes both aleatory and epistemic uncertainties of the basic variables. The HLWRS-ISG-01 (NRC 2006, p. 2) states that the simplified approaches in Chapter 4 of EPRI TR-103959 (EPRI 1994) are acceptable; therefore, a method to combine the basic variable beta values (EPRI 1994, p. 4-6) is shown in Equation 4-12:

$$\beta = \left(\sum_i \beta_i^2\right)^{1/2} \qquad \text{(Eq.4-12)}$$

In this equation, *i*, indicates the ith underlying basic variable. The formula for each standard deviation, β_i, is:

$$\beta_i = \frac{1}{|\phi|}\ln\left[\frac{SF^i_{\phi\sigma}}{\overline{SF}}\right] \qquad \text{(Eq. 4-13)}$$

In Equation 4-13, ϕ is the number of standard deviations for which a lower scale factor is to be estimated. It is usually set to 1 for response variables or -1 for strength variables. $SF^i_{\phi\sigma}$ is the scale factor calculated by performing a stress calculation using the ith basic variable value that corresponds to ϕ standard deviations from the median. $\overline{SF}$ corresponds to the scale factor where all basic variable values in the stress calculation are at their median values.

Implementation of Equations 4-11, 4-12, and 4-13 proceeds as follows:

- Calculate the scale factor $\overline{SF}$ using all basic variable inputs at their median values.
- Equation 4-11 then produces the median capacity
- Repeat the analysis by changing each of the basic variables (Section 4.4.4.2), one at a time, to their plus or minus one standard deviation value.
- Equation 4-13 then yields each β_i.
- Equation 4-12 then yields β.

With the median capacity and the above beta values, a composite mean fragility curve will be obtained.

4.4.4.2 Basic Variables for Calculating Equipment Fragilities

Equipment fragilities are sensitive to basic variables used in developing the in-structure response as well as basic variables specific to equipment. *Methodology for Developing Seismic Fragilities* (EPRI 1994) provides a discussion of both the structure and equipment specific basic variables important to equipment fragility calculations. It is the variation of these variables owing to epistemic and aleatory uncertainty that leads to a fragility curve.

4.5 EVENT SEQUENCE ANALYSIS

Systems analysis has been part of probabilistic seismic assessment since the 1970s (Kennedy et al. 1980). Development of initiating events proceeds using a master logic diagram through event sequence diagrams (ESD), event trees, fault trees, and quantification (NRC 1983). An ESD or event tree represents scenarios in terms of initiating events, pivotal events, and end states. The initiating event in an ESD or event tree may be the occurrence of an earthquake. Alternatively, the initiating event may be a system, subsystem, or equipment failure event depicted by a fault tree, whose events are caused by an earthquake. Pivotal events, sometimes called top events, represent the response of SSCs to the initiating event. End states for this analysis will be OK, indicating no adverse consequences of interest or radionuclide release indicating a potential for worker, onsite public, or offsite public dose.

For earthquake-induced failures, probabilities of the initiating and pivotal events are developed using the methods described in the Sections 4.3 and 4.4. Such failures are a function of the ground motion level, so ground motion hazard curves need to be considered. For active component random failures, conditional probabilities from the internal events analysis will be used. Event tree development and quantification will be implemented in specially-developed computer codes such as SAPHIRE (V. 7.26, STN: 10325-7.26-00).

Sometimes the events depicted in ESDs or event trees cannot easily be mapped to available information about the occurrence probability of the event. It is often necessary, therefore, to disaggregate or breakdown these events to a simpler level in order to create a mapping between ESD or event tree events and the available failure data. In risk assessment, the most common form of this mapping is a fault tree. Fault trees are reduced to minimal cutsets, which provide insight into combinations of events most important to system probabilities. Codes such as SAPHIRE link fault trees such that scenario minimal cutsets are obtained providing insight into scenario frequencies and forming the basis of importance rankings.

A subset of the scenarios depicted in an event tree will be event sequences that lead to a possible worker or public dose. These scenarios are event sequences as defined in 10 CFR 63.2. As is standard practice, the internal event trees and fault trees are used as the basis for developing the seismic event sequence analyses. The internal initiating events are replaced with the appropriate seismic initiating event. Structure and passive component events are added to the internal event fault trees as needed to complete the seismically induced scenarios. Furthermore, additional event trees may be developed that include SSCs not normally included in the internal events analysis. Fault trees in support of the event sequence quantification will include both seismically induced failures and coincident random failures, as is standard practice. Seismically induced failures require development of fragility curves as outlined in Section 4.4, and random failure probabilities are derived from the event sequence analysis performed for the internal events. The quantification of event sequences using fragility curves and coincident random event probabilities will be performed using an appropriate computer code (e.g., SAPHIRE). This results in a curve of conditional probability as a function of input motion. The unconditional mean event sequence probability results from application of the standard stress-strength interference integral (sometime referred to as a convolution) as shown in Equation 4.11.

$$P_f = \int_0^{a_{\max}} F(a) \left| \frac{dH(a)}{da} \right| da \qquad \text{(Eq. 4.11)}$$

where P_f is the mean failure probability, F(a) is the mean (sometimes called composite) cumulative distribution function over acceleration (i.e., fragility curve of an SSC), and H(a) is the mean hazard curve as a function of acceleration, which in this equation is also a cumulative distribution function. Note that, strictly H(a) is the mean annual rate of exceedance, and P_f is the mean annual rate of failure. For small numbers, these are approximately equal to the annual probabilities of exceedance and failure.

The unconditional frequency of each event sequence will be compared against the Category 1 and Category 2 probability thresholds. Those event sequences below the Category 2 threshold will be screened out. Event sequences, such as those including major building structures, may be shown to be screened out without quantifying the entire event sequence. However, complete event sequence descriptions will be developed.

4.6 PERFORMANCE EVALUATION

4.6.1 Development of Event Sequences

The risk-informed performance based process of design and analysis given in this document will lead to convergence on a design that meets the performance objectives of 10 CFR Part 63. If necessary, the design basis of SSCs within event sequences that results in potential consequences in excess of Category 1 dose limits (Table 3-1) will be modified such that the probability of the event sequence will be less than unity over the preclosure period. In application of the methodology, seismically initiated event sequences, which have a probability less than unity before permanent closure of the geologic repository, are evaluated individually in accordance with 10 CFR 63.111(b)(2), and the sequence probabilities are not aggregated as in a complete probabilistic risk assessment. The design basis of SSCs, which are initially part of Category 2 event sequences, will be modified (if necessary) such that either an estimated dose is less than the 10 CFR 63 requirement or the probability is reduced to below the Category 2 lower probability threshold. Each event sequence will be qualitatively developed from an initiating event through the potential for a release of radionuclides. Quantification of each event sequence will begin with the initiating event and extend through the pivotal events until either an end state is reached or the probability is shown to be below the Category 2 threshold. Mean values will be developed for purposes of quantification, which, by their development, incorporate applicable uncertainties. The remaining pivotal events will be conservatively assigned a failure probability of unity. This approach provides a visual depiction of margin between the threshold and the calculated sequence probability. It has been shown that mean values are appropriate for purposes of deciding if a goal is met (Howard 1988, p.91-98, pp. 91-98). The NRC staff guidance recommends the use of mean values to assess compliance (NRC 2006).

4.6.2 Annual Probability of the Category 2 Performance Objective

For quantitative probabilistic evaluations, the probability of each event sequence will be evaluated in terms of annual probability of occurrence. To compute the Category 2 performance goal (i.e., at least 1 chance in 10,000 over the preclosure period) in terms of annual probability, consideration must be given to both the duration of the preclosure period and the period over which a hazard is expected to be present. In particular, the repository may have selected facilities licensed to function for periods less than the total preclosure period and this needs to be considered when converting the Category 2 performance goal in terms of annual probability.

To illustrate, for an SSC that performs a safety function during the entire preclosure period, and assuming a preclosure period of 100 years, the Category 2 performance goal becomes (1/10,000) ÷ 100 yrs = 10^{-6}/yr. If the preclosure period is shortened, the goal becomes larger; (e.g., with a 25-yr preclosure period, the Category 2 performance goal becomes: (1/10,000) ÷ 25 yrs = 4×10^{-6}/yr).

The relationship also changes if the SSC is not expected to perform a safety function for the entire preclosure period. Again, assuming a preclosure period of 100 years, but with an SSC that is expected to perform a safety function half of the preclosure period (e.g., because of the limitation of surface operations to no more than 50 years), the Category 2 performance goal becomes (1/10,000) ÷ 50 years = 2×10^{-6}/yr.

5. MITIGATION OF FAULT DISPLACEMENT HAZARDS

This section describes the methods, procedures, and criteria that the DOE intends to use to provide reasonable assurance that ITS SSCs will meet the pertinent 10 CFR Part 63 preclosure performance objectives with respect to fault displacement. The primary design approach for fault displacement is to locate (whenever feasible) ITS SSCs away from Quaternary faults with a potential for significant displacement so that no explicit fault displacement design is required. NUREG-1451 (McConnell et al. 1992, Section 3.1.3) defines "Type I" faults as faults or fault zones that are subject to displacement and that may affect repository design and/or performance. This definition includes two components: (1) “subject to displacement” implies that the fault is a Quaternary fault, and (2) “may affect the design and/or performance” implies that an evaluation has been made of design and/or performance significance. Without *a priori* knowledge of design and/or performance significance, one cannot indicate which faults are Type I and which are not. As a result, the terminology “Quaternary fault with potential for significant displacement” is used in this document to indicate those faults that may potentially be Type I faults.

As discussed in Section 5.2, if fault displacement is considered in the design of an SSC, the design basis fault displacement hazard levels shall be a factor of 10 lower MAPE than those for ground motions. A description is given of the approach to determining design basis fault displacements and the fault-displacement design acceptance criteria.

5.1 CRITERIA FOR FAULT DISPLACEMENT HAZARD AVOIDANCE

Unlike vibratory ground motion hazard, fault displacement hazard is concentrated at the location of faults. Consequently, the exposure of SSCs to fault displacement hazard can be limited by avoiding the locations of faults that have a significant potential for fault displacement. Fault avoidance is the DOE preferred approach to mitigating fault hazards. Whether the potential for fault displacement is significant depends on the SSC in question. The hazard is judged significant when an explicit fault displacement design might be necessary to accommodate the hazard. Conversely, the hazard is judged negligible—and fault displacement hazard avoidance is achieved—when the amplitude of displacement is so low that there clearly is no need for the SSC in question to have an explicit fault displacement design.

Given the variability and uncertainties regarding the amount and recurrence rate of displacement episodes on local faults, fault displacement hazard has been assessed probabilistically (CRWMS M&O 1998, Section 8), and design basis fault displacements are expressed as fault displacement hazard curves at particular demonstration sites. For example, the fault displacement hazard curve for the Midway Valley site, within which the surface facilities are located, is given in *Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada* (CRWMS M&O 1998, Figure 8-14). To account for the uncertainties and acknowledging less experience in seismic design for displacement than for vibratory ground motions, the design basis probability levels are one order of magnitude lower than for ground motions. The MAPE is 10^{-4} for design basis fault displacement (DBFD)-1 and 5×10^{-5} for design basis fault displacement DBFD-2. The DOE criteria for fault displacement hazard avoidance are consistent with these exceedance probabilities. Specifically, the DOE will assess the probabilistic fault displacement hazard for each ITS SSC, identify the fault displacement that corresponds to the applicable DBFD level, and judge whether, for that displacement level, an explicit fault displacement design would be necessary. Fault displacement hazard avoidance is achieved if the level is low enough that an explicit fault displacement design is not necessary.

The DOE expects that fault displacement hazard avoidance can be achieved for all ITS surface and subsurface SSCs that are spatially compact. Fault displacement hazard avoidance may or may not be feasible for all subsurface SSCs that are spatially extended. In any case, if fault displacement hazard avoidance is not feasible for any SSC that is ITS, then it will be designed to accommodate the applicable design basis fault displacement without loss of safety function.

It is possible that spatially extended SSCs that cross Quaternary faults with potential for significant fault displacement will be classified as ITS. If this is the case, the DOE will use the probabilistic fault displacement hazard analysis (CRWMS M&O 1998), identify the fault displacement that corresponds to the applicable DBFD-1 or DBFD-2 MAPE at the particular location of the SSC, and determine whether an explicit fault displacement design is necessary to accommodate the potential displacement with high confidence. If a fault displacement design is required, it will be executed per the acceptance criteria described in Section 5.2.2.

5.1.1 Implementation of NRC Staff Technical Position on Consideration of Fault Displacement Hazards in Geologic Repository Design

The NRC staff position on the consideration of fault displacement hazards in geologic repository design is published in NUREG-1494 (McConnell and Lee 1994, Section 3(2)). NUREG-1494 states that “In general, areas within the controlled area of a geologic repository that contain "Type I" faults should be avoided, where this can be reasonably achieved, when locating structures, systems, and components important to safety or important to waste isolation.”

As described in Section 5.1, the DOE approach to fault avoidance is, where feasible, to locate ITS SSCs where the fault displacement hazard is so low that no explicit fault displacement design is required. This approach inherently avoids Quaternary faults with potential for significant displacement and is consistent with the staff guidance in NUREG-1451 (McConnell et al. 1992).

The assessment of fault displacement (and vibratory ground motion) hazards at the site was conducted using the methodology described in the first seismic topical report (YMP 1997a) and is documented in *Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada* (CRWMS M&O 1998). To support the seismic hazard assessment, the DOE has mapped in detail the faults at and near the site and has investigated all known and suspected Quaternary faults in the Yucca Mountain region that are of sufficient length and located such that they could materially contribute to the vibratory ground motion or fault displacement hazard at the site. The DOE concludes that the level of detail in the fault investigations and the area investigated has been sufficient to identify all Quaternary faults with potential for significant displacement that could impact preclosure repository design. The probabilistic fault displacement hazard analysis (CRWMS M&O 1998, Section 8) provides hazard curves that will be used to assess the amount of fault displacement associated with the DBFD-1 and DBFD-2 annual probability levels at any particular SSC location within the geologic repository operations area.

5.1.2 Compliance with Preclosure Performance Objectives

The fundamental approach to addressing fault displacement is the avoidance of Quaternary faults with the potential for significant displacement. If such faults cannot be avoided, then the DBFD levels given in Section 5.2.1 should be used for design (DBFD-1 or DBFD-2). This approach will provide reasonable assurance that the pertinent 10 CFR 63.111 preclosure performance objectives have been met with respect to fault displacement.

10 CFR 63.102(f) recommends limiting initiating events for event sequence analysis based on precedence for nuclear facilities having comparable or higher levels of risk significance. Regulatory Guide 1.165, Appendix A, which is used for nuclear power plants, defines a capable tectonic source (fault) by the following:

> **Capable Tectonic Source**—A capable tectonic source is a tectonic structure that can generate both vibratory ground motion and tectonic surface deformation such as faulting or folding at or near the earth's surface in the present seismotectonic regime.
>
> It is described by at least one of the following characteristics:
>
> a. Presence of surface or near-surface deformation of landforms or geologic deposits of a recurring nature within the last approximately 500,000 years or at least once in the last approximately 50,000 years.
>
> b. A reasonable association with one or more moderate to large earthquakes or sustained earthquake activity that are usually accompanied by significant surface deformation.
>
> c. A structural association with a capable tectonic source having characteristics of either section a or b in this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other.

Notwithstanding the potential implications of other evidence given in paragraphs b and c, the regulatory guidance leads to the conclusion that tectonic sources (faults) that have not undergone displacement in the past 500,000 years are considered to be "not capable" and do not need to be considered further for seismic hazard analysis. Accordingly, use in this topical report of the Quaternary time period (past approximately 1.8 million years) to define faults of significance is reasonable and conservative relative to nuclear power plant precedent. Therefore, it is appropriate to limit the initiating event for fault-displacement event sequences to those faults that have evidence of Quaternary displacement.

Per the definition of Category 2 event sequences given in 10 CFR 63.02 (one chance in 10,000 during the preclosure period), the threshold for screening out these event sequences is an annual frequency or probability of less than 2×10^{-6} for a 50-year preclosure period. A fault displacement event that has not occurred in 1.8×10^{6} years could be assessed to have a probability of occurrence that is approximately $1/(1.8 \times 10^{6}) = 5.6 \times 10^{-7}$ or less. This provides a basis for concluding that the event sequences initiated by a faulting event on a pre-Quaternary fault can be screened out from further consideration on the basis that the initiating event itself has a probability of less than 10^{-6}.

5.2 CRITERIA FOR FAULT DISPLACEMENT DESIGN

5.2.1 Determination of Design Basis Fault Displacements

When fault displacement hazard avoidance is not feasible for ITS SSCs, these SSCs will be designed to withstand the DBFD without loss of their required safety functions. The DOE considers that probabilistic criteria for DBFDs are most appropriate to implement a risk-informed design process. Specifically, the DOE considers that fault displacements having mean annual exceedance probabilities of 10^{-4} and 5×10^{-5} are appropriate for DBFD-1 and DBFD-2, respectively. These values are a factor of 10 lower than the exceedance probabilities of the corresponding DBGM-1 and DBGM-2 reflecting the more limited experience with engineering design of facilities for fault displacement and with assessments of fault displacement hazards.

The motivation and justification for using a probabilistic rather than deterministic description of DBFDs are the same as for DBGMs. Specifically, the advantages are that a properly done probabilistic fault hazard analysis captures and reflects both the variability and uncertainty, and accounts for both the magnitude and the likelihood of occurrence of the hazard. Therefore, a probabilistic approach provides the information that is needed to implement a risk-informed design methodology, as well as providing the type of information that is needed for the PCSA in order to evaluate compliance with the preclosure performance objectives.

5.2.2 Acceptance Criteria for Fault Displacement Design

Fault displacement loads depend on the amount and direction of the fault movement and on the ease with which the two parts of the SSC on two sides of the fault can move relative to each other. The latter depends on:

- The stiffness (or flexibility) of the SSC or supporting structure, especially in the vicinity of the fault
- The stiffness (or flexibility) of the ground around the buried segment or foundation of the SSC, especially in the vicinity of the fault
- The configuration of the SSC.

Once the DBFDs are determined, the resulting loads (or stresses) and deformations (or strains) in the SSC will be calculated using analytical models that will consider the three parameters. When similar loads/stresses and deformations/strains are calculated for vibratory ground motion, it is customary to use stress-based acceptance criteria to establish design adequacy assuming essentially linear elastic behavior, which is the basis for industry codes and standards. Unlike vibratory ground motion loads, however, fault displacement loads are generally localized and often cause inelastic response of SSCs, unless the SSC and the ground medium are very flexible, in which case the SSC can undergo large deformation and stay within elastic limits. For this reason, the DOE intends to use strain-based acceptance criteria to establish the design adequacy of SSCs subjected to fault displacement loads.

In establishing strain-based acceptance criteria for the Yucca Mountain repository facilities, nuclear power plant and industry experiences with the use of strain-based criteria will be used. Examples are the strain criteria used for designing pipe rupture restraint systems and for designing SSCs subject to accidental impact and impulse loads such as those resulting from tornado missiles, turbine missiles, aircraft crashes, cask drops, reactor vessel head drops, and others that may be applicable. Some similarities also exist between localized inelastic response of SSCs when subject to fault displacement loads and localized stress well beyond linear elastic limit of materials. Because of uncertainties in the fragilities of SSCs however, the design acceptance criteria for fault displacement loads will not permit strain levels up to the ultimate or failure strain limit of the material. Instead, the limiting strain will be determined by considering the parameters that influence the fragility of the SSC. Explicitly, these are the configuration of the SSC, the SSC failure mode, the SSC material characteristics (brittle versus ductile), the stiffness of the SSC, and the stiffness of the ground material near the fault. Considering these parameters, strain limits will be established on a case-by-case basis to provide reasonable assurance that the seismic safety goal established for the SSC will be achieved.

In addition to imposing strain-based acceptance criteria when an explicit fault displacement design is required, the DOE will follow conservative layout guidelines when locating ITS SSCs relative to Quaternary faults with potential for significant displacement. For instance, when practical layout requirements make it necessary to place spatially extended SSCs across a Quaternary fault with potential for significant displacement, the layout will be configured such that the SSC crosses the fault trace at a steep angle, minimizing the exposure of the SSC to faulting-induced damage.

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6. SUMMARY

This topical report describes the methodology and criteria that the DOE intends to use for preclosure seismic design of SSCs that are ITS and credited with preventing or mitigating the consequences of a seismically initiated event sequence in the geologic repository operations area. This report also describes a methodology using probabilistic seismic analyses for demonstrating compliance to the performance objectives in 10 CFR 63.111(b)(2). The establishment of preclosure seismic design criteria involves both PCSA as well as repository design. Evaluations of preclosure safety are made with respect to a reference design for the geologic repository operations area. Therefore, the fundamental SSCs of the preclosure design are evaluated in the PCSA to identify those SSCs that are ITS, per the definition given in 10 CFR 63.2. This report provides a risk-informed methodology for assigning seismic DBGMs to those SSCs that have been determined to be ITS and credited with preventing or mitigating the consequences of a seismically initiated event sequence.

Seismic safety is achieved through a combination of two important design aspects: (1) the DBGM level, and (2) the conservatism in the design codes, standards, and acceptance criteria as found in NUREG-0800 (NRC 1987, Chapter 3)). Per regulation, the DOE has the flexibility to choose whatever seismic design bases and design procedures it feels will provide reasonable assurance that the preclosure performance objectives of 10 CFR Part 63 are met. The use of appropriate levels of DBGMs coupled with the adoption of the nuclear power plant seismic design codes, standards, and acceptance criteria identified in NUREG-0800 (NRC 1987, Chapter 3) are part of the design methodology described in this topical report.

Two DBGM levels will be used for the seismic design of ITS SSCs:

- DBGM-1 with a MAPE = 10^{-3} (1,000- year return period)
- DBGM-2 with a MAPE = 5×10^{-4} (2,000- year return period).

The determination of appropriate DBGM levels for specific SSCs depends on their risk significance (i.e., radiological consequences). The ITS SSCs identified in seismically initiated event sequences will be identified in the PCSA and, depending on the radiological consequences of the event sequences, will be assigned DBGM-2 or DBGM-1.

NUREG-0800 (NRC 1987, Chapter 3) identifies the methods and procedures that are applicable to the seismic design of nuclear power reactors and identifies specific acceptance criteria, regulatory guides, and industry standards that provide information, recommendations, and guidance for compliance. With the exceptions identified in this document, the DOE considers that specific codes, standards, and acceptance criteria provided by NUREG-0800 (NRC 1987, Chapter 3) are appropriate for use in preclosure seismic design.

To ensure that the combination of DBGMs and design codes, standards, and acceptance criteria are adequately conservative, seismic margin assessments will be conducted for the major structures. The seismic margins assessment will show that the HCLPF capacity of the major structures will exceed the BDBGM event (review level earthquake). This will ensure that adequate seismic design margins will exist for these structures, such that they will maintain their defined functions credited in the PCSA. Probabilistic seismic analyses will be conducted to demonstrate compliance with the preclosure performance objectives in 10 CFR 63.111(b). The key components of the probabilistic seismic analyses include: (1) development of mean hazard curves for pertinent ground motion measures at MAPEs below 10^{-6}, (2) development of fragility curves for specific ITS SSCs credited in event sequences, (3) development of seismically initiated event sequences, (4) evaluation of the dose consequences of the seismically initiated event sequences, (5) categorization as Category 1 or Category 2 event sequence per 10 CFR 63.2, (6) convolution of seismic hazard curves and fragility curves, and (7) assessment of probabilities of event sequences. For each seismically-initiated event sequence, the probabilistic seismic analyses will demonstrate that either:

- the annual probability of the seismic event sequence is less than one in 10,000 during the preclosure period, such that the event sequence may be screened out, or
- the radiological dose consequence of each Category 2 event sequence that is not screened out meets the performance objectives of 10 CFR 63.111(b)(2)

This document also describes the methods, procedures, and criteria that the DOE intends to use to provide reasonable assurance that ITS SSCs will meet the pertinent 10 CFR Part 63 preclosure performance objectives with respect to fault displacement. The primary design approach for fault displacement is to locate (whenever feasible) ITS SSCs away from Quaternary faults with potential for significant displacement so that no explicit fault displacement design is required. However, for those SSCs (if any) that must consider fault displacement, the DBFD levels shall be a factor of 10 lower MAPE than those for ground motions.

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APPENDIX A
LOW-RISE CONCRETE SHEAR WALL STRUCTURES

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APPENDIX A
LOW-RISE CONCRETE SHEAR WALL STRUCTURES

A-1. INTRODUCTION

This appendix describes the details for development of the high confidence capacity and permissible drift limits of low-rise concrete shear walls.

A-1.1 HIGH CONFIDENCE SHEAR STRENGTH CAPACITY OF LOW-RISE CONCRETE SHEAR WALLS

A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1) (EPRI 1991, pp. 2-51 to 2-53) recommends that the 98% exceedance probability capacity can be estimated by defining the material strengths at the 95% exceedance probability and the capacity equations at the 84% exceedance probability. Code-established material strengths (i.e., code-specified yield and ultimate strengths of steel and concrete) are specified at about the 95% exceedance probability or higher. Therefore, if these code-specified material strengths are used, it is sufficient to establish the capacity equation at the 84% exceedance probability.

A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1) (EPRI 1991, Appendix L) shows that the nominal (median) ultimate shear strength, v_n, of a low-rise concrete shear wall with height, h_w, less than twice its length, l_w, is given by:

$$v_n = 8.3\sqrt{f_c'} - 3.4\sqrt{f_c'}\left(\frac{h_w}{l_w} - 0.5\right) + \frac{N_A}{4 l_w t_n} + \rho_{se} f_y \qquad \text{(Eq. A-1)}$$

where

f_c' = concrete strength

N_A = axial load on the wall (compression positive)

t_n = nominal wall thickness

f_y = reinforcing steel yield strength

ρ_{se} = effective steel reinforcement ratio, which can be expressed as:

$$\rho_{se} = A\rho_v + B\rho_h \qquad \text{(Eq. A-2)}$$

where ρ_v and ρ_h are the fraction of vertical and horizontal steel, respectively, and for:

$$\begin{array}{lll} h_w/l_w \le 0.5 & A = 1 & B = 0 \\ 0.5 \le h_w/l_w \le 1.5 & A = -h_w/l_w + 1.5 & B = h_w/l_w - 0.5 \\ 1.5 \le h_w/l_w & A = 0 & B = 1 \end{array} \qquad \text{(Eq. A-3)}$$

A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1) (EPRI 1991, Appendix L, pp. L-3 and L-4) also shows that the logarithmic standard deviation, β, of data about this median capacity is about 0.20. Thus, the 84% exceedance probability ultimate shear capacity, v_u, is:

$$v_u = \phi v_n \qquad \text{(Eq. A-4)}$$

where $\phi = e^{-\beta} = 0.80$ is the required strength reduction factor.

Equation A-4 defines the high confidence shear strength capacity of a low-rise concrete shear wall if code-specified minimum values are used for f_c' and f_y in Equation A-1. Additional details are given in *A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)* (EPRI 1991, Appendix L). This approach can be used to estimate the high confidence strength capacity for other failure modes if adequate test data are available.

A-1.2 PERMISSIBLE DRIFT LIMITS FOR LOW-RISE CONCRETE SHEAR WALLS

Based on an extensive review of test data for shear walls subjected to cyclic loads, *Shear Wall Ultimate Drift Limits* (Duffey et al. 1994) established estimates of the shear drift capability of concrete shear walls.

Duffey et al. (1994, Figure 2.1) show that the drift limit corresponding to ultimate load capacity is insensitive to aspect ratio (h_w/l_w) for aspect ratios less than about 1.0. For higher aspect ratios, the drift limit increases with increasing aspect ratios. Therefore, Duffey et al. (1994, p. 16) established drift limit recommendations for concrete shear walls with aspect ratios between 0.24 and 1.07. Extending these drift limits to higher aspect ratio walls is conservative.

Test data also shows that the drift limits corresponding to ultimate load capacity for walls with aspect ratios less than 1.0 increase when a large percentage of vertical reinforcing steel is present. Therefore, the results presented by Duffey et al. (1994, p. 16) are limited to walls with small percentages of vertical reinforcing steel ranging from 0.0% to 0.86% with a median ratio of about 0.5%. Therefore, the use of these drift limit results for walls with higher reinforcing steel percentages is conservative.

Even low-rise concrete shear walls with low steel percentages can drift about 2% of their story height before they will actually fail. However, under cyclic loading, their strength capacity will degrade on subsequent cycles when prior drifts exceed certain lesser limits. Therefore Duffey et al. (1994, Table 4.3) presented drift limits corresponding to retention of load capacity ranging from 100% to 50% of the original ultimate capacity for subsequent cycles. Duffey et al. (1994, Table 4.3 and Section 4.2) summarized the median drift limits, $DL_{50\%}$, and logarithmic standard deviations, β, on these drift limits as a function of ultimate load capacity retained during subsequent nonlinear cycles. This information is repeated in Table A-1. The 95% confidence drift limits, $DL_{95\%}$, are given in Table A-1 as computed from:

$$DL_{95\%} = DL_{50\%} e^{-1.645\beta} \qquad \text{(Eq. A-5)}$$

where 1.645 is the standardized normal variant associated with the 5% exceedance probability (modified from Kennedy (2001, p. 40)).

The Limit State C drift limit of 0.4% for low-rise concrete shear walls corresponds to a drift at which there is about 95% confidence that the ultimate load capacity will be retained during subsequent load cycles (DOE-STD-1020-2002, Section 2.3). Therefore, there is about 95% confidence that the wall will retain its full strength and will remain fully serviceable. Although not shown in Table A-1, test data indicate that cracks will remain small (DOE-STD-1020-2002, Table C-2).

Similarly, the Limit State A drift limit of 0.75% for low-rise concrete shear walls corresponds to a drift at which there is about 95% confidence that about 50% of the ultimate load capacity will be retained, and about 50% confidence that 100% of the ultimate load capacity will be retained. This retained strength is sufficient to provide high confidence that collapse will be prevented. However, the structure might suffer significant damage and might not be repairable back to its original strength.

The Limit State B drift limit of 0.6% corresponds to a drift at which there is about 95% confidence that 80% of the ultimate load capacity will be retained. Test data show that crack widths might begin to become significant beyond this drift limit. These larger cracks could reduce the capacity of anchorage that anchors components to the concrete.

Table A-1. Drift Limits for Low-Rise Concrete Shear Walls as a Function of Percentage of Ultimate Load Capacity Retained During Cyclic Loading [a]

Percentage of Ultimate Load Retained (%)	Median Drift Limit ($DL_{50\%}$)(%)	Logarithmic Standard Deviation (β)	95% Confidence Drift Limit ($DL_{95\%}$)(%) [b]	Corresponding Limit State [c]
100	0.72	0.373	0.39	C
90	1.00	0.437	0.49	–
80	1.24	0.452	0.59	B
70	1.48	0.464	0.69	–
60	1.64	0.524	0.69	–
50	1.84	0.566	0.73	A

NOTES: [a] Table modified from Duffey et al. 1994, Table 4.3.
[b] Computed using EQ. A-5.
[c] Limit states identified in text.

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APPENDIX B
ACRONYMS AND ABBREVIATIONS

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APPENDIX B
ACRONYMS AND ABBREVIATIONS

BDBGM	beyond design basis ground motion
CDFM	Conservative-Deterministic-Failure-Margin
DBFD-1, -2	Design Basis Fault Displacement-1, -2
DBGM-1, -2	Design Basis Ground Motions-1, -2
DE	design earthquake
DOE	U.S. Department of Energy
ESD	event sequence diagram
GROA	geologic repository operations area
HCLPF	high confidence of low probability of failure
HLW	high-level radioactive waste
IPEEE	Individual Plant Examination of External Events
ISA	Integrated Safety Analysis
ISFSI	independent spent fuel storage installation
ISG	interim staff guidance
ITS	important to safety
LA	license application
MAPE	mean annual probability of exceedance
mrem	millirem (10^{-3} rem)
MRS	monitored retrievable storage
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
PC-3	Performance Category 3
PCSA	preclosure safety analysis
PFS	Private Fuel Storage, L.L.C.
PGA	peak ground acceleration
PSHA	probabilistic seismic hazard analysis
rem	roentgen equivalent man; a unit for measuring absorbed doses of radiation

APPENDIX B
ACRONYMS AND ABBREVIATIONS (CONTINUED)

SMA	seismic margin assessment
SSCs	structures, systems, and components
SSE	safe shutdown earthquake
STR#1	first seismic topical report (*Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain)*
STR#2	second seismic topical report (*Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain*)
STR#3	third seismic topical report (not issued)
S_v	Sievert; unit of radiation dose equivalent (1 Sv equals 100 rem)
TEDE	total effective dose equivalent
UHS	uniform hazard spectra
YMP	Yucca Mountain Project
YMRP	*Yucca Mountain Review Plan, Final Report*