U.S. NUCLEAR REGULATORY COMMISSION

PRE-DECISIONAL
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Technical Lead: Jim Xu

TECHNOLOGY-INCLUSIVE, RISK-INFORMED, AND PERFORMANCE-BASED METHODOLOGY FOR SEISMIC DESIGN OF COMMERCIAL NUCLEAR PLANTS

A. INTRODUCTION

Purpose

This pre-decisional, draft Regulatory Guide (pre-decisional DG) provides the U.S. Nuclear Regulatory Commission (NRC) staff's initial preliminary draft regulatory guidance for applicants who seek to use a technology-inclusive (TI) and risk-informed and performance-based (RIPB) approach for the seismic design in preparing an application for a license for a commercial nuclear plant under Section 53.480, "Earthquake Engineering," in Subpart C, "Design and Analysis Requirements," of 10 CFR Part 53, "Risk-informed, technology-inclusive regulatory framework for commercial nuclear plants." The NRC staff plans to prepare a DG for public comment based on this pre-decisional DG and will consider discussion of the pre-decisional DG at ACRS meetings, ACRS views on the pre-decisional DG, and stakeholder input from public meetings and public comments.

Applicability

A Regulatory Guide (RG) developed from this pre-decisional DG would be applicable to designers, applicants, and licensees of commercial nuclear power plant applying for permits, licenses, certifications, and approvals under Framework A of 10 CFR Part 53, "Risk-informed, technology-inclusive regulatory framework for commercial nuclear plants."

Applicable Regulations

- Preliminary Proposed 10 CFR Part 53, Framework A, Subpart C, § 53.480, "Earthquake Engineering" provides regulatory requirements for engineering activities related to seismic design and analyses required for meeting Part 53 Framework A safety criteria as well as shutdown of an operating reactor following an earthquake event.

- Preliminary Proposed 10 CFR Part 53, Framework A, Subpart D, § 53.500, "General siting" § 53.510 "External hazards," and § 53.520, "Site characteristics" provide siting requirements for site geologic, geophysical, geotechnical, and seismological investigations to characterize the seismic hazard and develop site specific earthquake ground motions. These earthquake ground motions are characterized by the design basis ground response spectra (DBGMs) that are used for performing the engineering design under 10 CFR Part 53, Framework A, Subpart C, § 53.480.

Related Guidance

- RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors" (Ref. 1), provides a risk-informed and performance-based (RIPB) framework for establishing the licensing basis for non-Light Water Reactors (LWRs).

- Trial RG 1.247, "Acceptability of Probabilistic Risk Assessment Results for Advanced Non-Light Water Reactor Risk-Informed Activities," (Ref. 2), describes an approach for determining whether a design-specific or plant-specific probabilistic risk assessment (PRA) used to support an application is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for non-LWRs.

- RG 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 3), provides an acceptable approach for determining whether a base PRA, in total or in the portions that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for LWRs.

- The NRC interim staff guidance (ISG), "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," issued December 2021 (Ref. 4), provides staff positions on a roadmap for advanced reactor applications.

- RG 1.208, "A Performance-Based Approach To Define The Site-Specific Earthquake Ground Motion" issued March 2007 (Ref. 5), provides an acceptable alternative for use in satisfying the requirements set forth in Section 100.23, "Geologic and Seismic Siting Criteria," of Title 10, Part 100, of the Code of Federal Regulations (10 CFR Part 100), "Reactor Site Criteria"

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. Regulatory guides are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if supported by a basis for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

[The Paperwork Reduction Act statement and public protection notice will be added to this location when this pre-decisional draft guidance is finalized in a draft guidance.]

Public Protection Notification

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B. DISCUSSION

Reason for Issuance

This pre-decisional DG provides technology-inclusive guidance for the seismic design of structure, system, and components (SSCs) for commercial nuclear plants for applications submitted under 10 CFR Part 53, Section 53.480, "Earthquake Engineering." The Nuclear Energy Institute (NEI) proposed an RIPB regulatory approach for selection of Licensing Basis Events (LBEs); safety classification of structures, systems, and components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy for non-light water reactors in Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," issued August 2019 (Ref. 6). The NRC endorsed NEI 18-04 in RG 1.233, with clarifications. RG 1.233 and NEI 18-04 adopt an RIPB framework, which defines a set of design-basis external hazard levels (DBEHLs) that form an important part of the design and licensing bases, which can be used under Part 53, Framework A.

The NRC uses these DBEHLs to determine the design-basis seismic events and other external events that the safety-related (SR) SSCs will be required to withstand under Part 53. The input to the seismic design of SSCs for commercial nuclear plants comes from the probabilistic seismic hazard analysis performed to characterize the site seismological conditions as part of the siting review, per the proposed § 53.510(c) of Subpart D to Part 53. For applications submitted under Framework A of Part 53, § 53.480, a variety of approaches could be used to support the seismic design from the traditional deterministic seismic design practice with adequate PRA-based margin to more recent risk-informed and graded alternatives similar to the effort by the Department of Energy and standards development organizations or extending to the full integration of seismic PRAs into the design and licensing of commercial nuclear plants. The § 53.480 options available to support the seismic design of plant SSCs also involves options in how to characterize the seismic hazards associated with a plant site. A detailed description of the relationship between the probabilistic seismic hazard analysis required in § 53.510(c), the design basis ground motions addressed in § 53.480 and 53.510, and their connections to the engineering design per § 53.480 is provided below in the sub-section titled "Relationship Between Proposed § 53.480 and Proposed Options 2 and 3".

There are three major steps involved in the seismic design process of a commercial nuclear plant: 1) seismic design criteria for determining required SSC capacities (typically relying on staff endorsed consensus nuclear codes and standards); 2) safety evaluation of the seismic design (e.g., SPRA); and 3) safety criteria required by the regulation that the SSCs through their seismic design need to meet. The intent of this pre-decisional DG is to provide various RIPB approaches for the seismic design process of SSCs of a nuclear plant by proposing three design options. Figure 1 shows these three options schematically. As discussed in later sections, Option 1 proposes to use as reference the current seismic design approach for LWRs with the corresponding guidance for its implementation.[1] This approach could be considered for the seismic design of both LWRs and non-LWRs, to meet §§ 53.480 and 53.510, since the technical aspects of the approach are not germane to any particular reactor design and therefore are appropriate for consideration under Part 53 Framework A. The current seismic design approach uses the PRA-based margin as an alternative to a seismic PRA for evaluating the design against the safety criteria (SECY 93-087 [Ref. 7] and ISG-20 [Ref. 8]) and has been applied to several new reactor applications (AP1000, ESBWR, Vogtle, etc.). Detailed guidance and criteria provided in Chapters 2, 3, and 19 of the Standard Review Standard Review Plan (SRP) could be adopted for Option 1 and therefore it is not discussed in detail in this pre-decisional DG.

In alternative proposed approaches, herein called Option 2 and Option 3, the seismic design approach that may be used to meet § 53.480 using the ground motion inputs as established by §53.510 is based on and integrated with the RIPB framework. Option 2 follows the framework for risk evaluation of the design as described in NEI 18-04, Revision 1, that is clarified in RG 1.233; while Option 3 resorts to traditional SPRA applied in a graded manner as endorsed by the staff for the risk assessment of the design.

The NEI 18-04 provides a technology-inclusive framework that considers external hazards in terms of functional design bases and discusses the determination of special treatment requirements for SSCs, which could include ensuring that they remain functional during and following seismic events. However, NEI 18-04 does not explicitly address how to incorporate seismic performance criteria into the physical design of SSCs. This issue related to seismic design of SSCs is discussed in NRC Research Information Letter (RIL) 2021-04, "Feasibility Study on a Potential Consequence-Based Seismic Design Approach for Nuclear Facilities," issued April 2021 (Ref. 9), which explores a seismic design approach

1 Option 1 proposes using established guidance and criteria similar to those afforded by Standard Review Plans (SRP, NUREG-0800), Chapters 2, 3, and 19. Therefore, this pre-decisional DG contains no technical positions related to Option 1.

consistent with the licensing modernization project (LMP) framework described in NEI 18-04, using the seismic performance criteria in ASCE/SEI 43-19.

Option 1 proposes to adopt the established seismic design approach based on SRP (NUREG-0800 [Ref. 10]) guidance for use by Part 53 applicants while proposed Options 2 and 3 are more risk-informed and provide greater flexibility in the design decisions using risk insights. Proposed Option 3 also includes the potential use of graded PRAs and other risk-informed analyses. Under this option, a blend of several variations of deterministic and probabilistic risk analyses could be considered based on their effect on risk-informed design decisions and commensurate with the broad spectrum of risk-informed approaches considered under § 53.480 in Subpart C of Part 53.

Background

The RIPB approaches currently used in evaluating seismic risks at operating reactors and other nuclear facilities include development of site-specific seismic ground motions using probabilistic seismic hazard analysis (PSHA) and seismic probabilistic risk assessments (SPRAs) to evaluate both plant-specific and generic issues. Currently accepted alternative RIPB regulatory approaches, such as risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors, are particularly germane to the approach in this pre-decisional DG. In the SSC categorization process, SSCs are categorized using Risk-Informed Safety Classes (RISCs) RISC–1, RISC–2, RISC–3, or RISC–4 for considerations for special treatment. The approach presented here extends the application of this concept to the seismic design. RIPB approaches for evaluating and addressing seismic risks is also at the heart of the flexibilities being included in § 53.480 in Subpart C of Part 53 (Framework A).

What distinguishes the process discussed in this pre-decisional DG from the established practice currently in use is the graded approach and design flexibility that are afforded by the direct incorporation of the RIPB concepts in the seismic design itself, as provided in § 53.480 with the ground motion inputs established in §53.510, so that the seismic hazard levels and design performance limits used in the seismic design are chosen for each SR SSC individually or in groups, commensurate with the SSC's contribution to risk and other safety and environmental considerations. Thus, in the RIPB seismic design process in this pre-decisional DG, the safety margins of individual SSCs are controlled according to their contribution to system-level and plant-level risk, thereby reflecting acceptable margins of safety that are realistic and achieve a more risk-balanced design.

The proposed Framework A in Part 53 has a hierarchy that includes defining safety criteria and providing safety functions needed to ensure that the safety criteria can be met both during normal operations and when a commercial nuclear plant experiences a licensing basis event. In turn, the safety functions are fulfilled by the design features and programmatic controls specified in later subparts in Framework A, including § 53.480 in Subpart C. Framework A proposes a performance-based approach, although it does include some prescriptive requirements. For example, the proposed § 53.480(c)(1) simply states that the safety-related and non-safety-related but safety significant SSCs must be able to withstand the effects of earthquakes commensurate with safety significance, without loss of capability to perform their safety function. In light of this flexibility, there are several ways to address design-basis ground motions for different categories of SSCs that could reflect their importance commensurate with safety significance and their capabilities to perform safety functions. This pre-decisional DG defines three options that the NRC staff proposes to find acceptable for meeting the requirements of § 53.480.

Proposed Option 1

To satisfy proposed § 53.480, Option 1 adopts the established seismic safety approach related to seismic hazards and SSC design with a conservative PRA-based safety margin. This approach is similar to the seismic design and evaluation process described in the SRP Chapters 2, 3, and 19, and related RGs, such as RG 1.208. Figure 1 shows the basic elements of Option 1. To implement Option 1 for applications under § 53.480 in Subpart C, a single design-basis ground motion (DBGM) is selected for all SR SSCs to be designed according to SDC-5 and limit state D (LS-D). To ensure that the seismic design based on Option 1 meets the proposed safety criteria in Part 53, Framework A, Subpart B, the seismic design for Part 53 applications would be carried out by adopting the acceptance criteria described in SRP Chapters 2 and 3, and the PRA-based seismic margin assessment (SMA) as an alternative to the SPRA for the risk evaluation of the design. The implementation is similar to that described in ISG-20 and SRP Chapter 19.

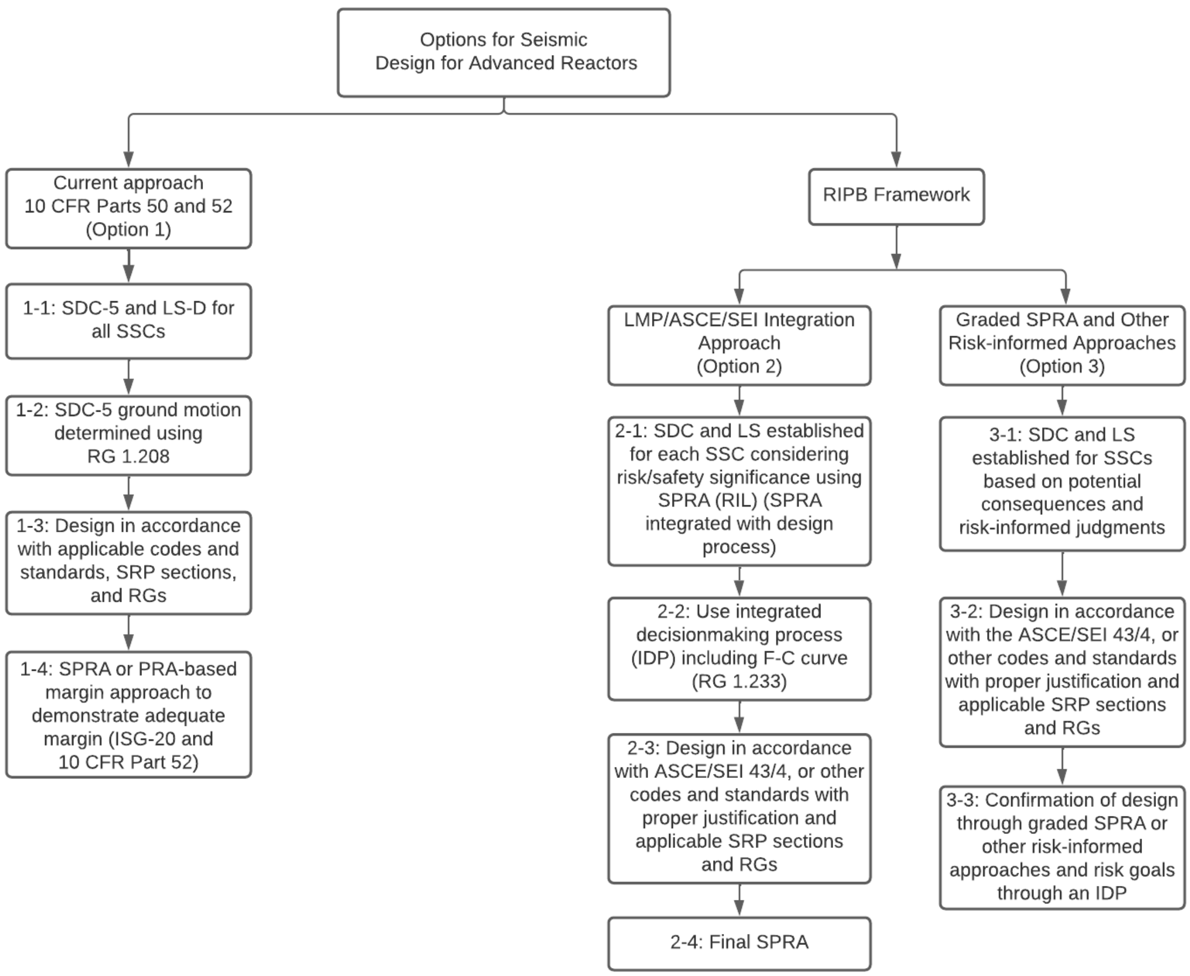

Figure 1. Three proposed options for advanced reactors seismic design

The SDC-5 ground motion has the smallest annual exceedance frequency (AEF) among all SDCs. Elastic design limits are specified for SSCs (although the current regulations do not require that only elastic limits be used, the practice and guidance are entirely built around elastic limits). The elastic design performance requirements correspond to LS-D in ASCE/SEI 43-19, the most stringent LS.[2] Boxes 1-1 and 1-2 of the flowcharts in Figure 1 allude to these elements of the current approach and requires use of LS-D. Box 1-3 refers to the actual design process of SSCs using the DBGM established in Box 1-2 and adopting the available regulatory guidance in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (Ref. 10). Box 1-4 represents the check of designs required in 10 CFR Part 52 through the SPRA or PRA-based margin approach.

In this context, proposed Option 1 aligns with ASCE/SEI 43-19, in that the DBGM associated with selecting SDC-5 for all SR SSCs is consistent with the design ground motions that are derived from the application of RG 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion” (Ref. 5). The design requirements for LS-D in nuclear codes and standards referenced in ASCE/SEI 43-19 are also similar to those provided in current NRC guidance and SRP sections.

This pre-decisional DG does not include specific technical positions for Option 1; since Option 1 adopts technical concepts and approaches related to seismic design from existing guidance. These seismic related technical approaches and concepts are not technology specific and could be adopted by Part 53 applicants. This pre-decisional DG also refers to aspects of the current guidance to provide overall context for the proposed options and their interrelationships.

Overview of Alternative Approaches (proposed Options 2 and 3)

The alternative seismic design approaches described in this pre-decisional DG for proposed Options 2 and 3 to address proposed § 53.480 in Subpart C of Part 53 provide a flexible set of design criteria for SSCs based on the various SDCs and allowable damage limit states (LSs) that are embedded in the most recent versions of industry codes and standards, such as ASCE/SEI 43-19 (Ref. 11) and ASCE/SEI 4-16, “Seismic Analysis of Safety-Related Nuclear Structures” (Ref. 12).

In addition to more flexible seismic design criteria, the alternative seismic design concept in RIL 2021-04 proposes using SPRA or other integrated plant analyses within the design process itself to provide plant designers with information on the seismic capacity required of individual SSCs and the overall plant, considering the role of each SSC in the plant design, construction, and operation.

The proposed RIPB framework in NEI 18-04 provides a timely opportunity to integrate the performance-based design approaches described in the above ASCE/SEI standards with the quantitative and qualitative risk and deterministic insights embedded in the RIPB framework. The LMP approach described in NEI 18-04 includes not only risk targets[3] (both for overall plantwide risk and for event sequences involving multiple SSCs) but also guidance for classifying SSCs that can account for the safety role of each SSC more directly. The LMP framework also places an emphasis on understanding individual event sequences (or groups of them) and an updated approach to assure defense-in-depth (DID). Finally, it more directly applies PRA modeling as a basis for much of the understanding that would support the safety decision-making process described in NEI 18-04.

The RIL 2021-04 explored an alternative concept to the current seismic design approach to integrate enhanced RIPB concepts within the LMP framework (proposed Option 2). This alternative concept is referred to as the “LMP/ASCE/SEI Integration Approach.” To illustrate the concept, Appendix

2 ASCE/SEI 43-19 specifies four LSs in terms of increasing structural deformations. LS-D is essentially elastic behavior with no permanent deformation after the design-basis earthquake. LS-C, LS-B, and LS-A allow deformations beyond elastic limits in increasing order.

3 As noted in RG 1.233, the frequency-consequence (F-C) target does not correspond to actual regulatory acceptance criteria but is a vehicle to assess a range of events to determine risk significance, support SSC classification, determine special treatment requirements, identify appropriate programmatic controls, and confirm the adequacy of DID.

A to this pre-decisional DG describes an example seismic design. This example aligns the LMP concepts with the performance targets described in ASCE/SEI 43-19 for seismic design. The RIL 2021-04 contains a detailed discussion of ASCE/SEI 43-19 and related standards. The ASCE/SEI 43-19 standard is a logical choice for many designs that use the LMP framework (or other RIPB frameworks) as their regulatory basis for licensing. ASCE/SEI 43-19 and ASCE/SEI 4-16 are performance-based standards for nuclear facilities. ASCE/SEI 43-19 contains a graded approach that allows for consideration of the risk significance of each component. These standards use well-known and well-practiced design procedures familiar to the nuclear industry. Adequate safety margins and risk-balance in a design are achieved through appropriate selection of DBGMs (each of the SDCs has an associated DBGM with a decreasing ground motion level from SDC-5 to SDC-3) and design LSs. Applicants can also incorporate standards other than ASCE/SEI 43-19 (e.g., ASCE/SEI 7-10, "Minimum Design Loads for Buildings and Other Structures" (Ref. 13) to design facilities with low risks.

More graded risk evaluation process of a proposed design than the LMP (e.g. RIL 2021-04) could potentially also be implemented under proposed Option 3. The basic approach would be to demonstrate that the adopted SDCs and LSs for each important SSC and the resulting overall design satisfy the intent of quantitative and qualitative risk goals and other considerations.

The integrated decision-making process (IDP)[4] described in NEI 18-04 and RG 1.233 can be used as an effective decision-making tool for the proposed Options 2 and 3. The IDP uses an RIPB integrated decision-making process, which is a structured, repeatable process by which decisions are made on significant nuclear safety matters, including consideration of deterministic and probabilistic inputs. The process is also performance-based because it employs measurable and quantifiable performance metrics to guide the determination of DID adequacy. For proposed Options 2 and 3, the IDP that would be in place to implement the methodology in NEI 18-04 is critical to selecting appropriate SDCs and LSs for SSCs. Conditions related to the IDP are included in Section C of this pre-decisional DG.

In both Options 2 and 3, it is important to realize that the risk-informed approaches allow for flexibility in assigning different DBGMs (i.e., SDCs) and LSs to SSCs and an evaluation of the risk contributions of the SSCs at the system or plant level. Once these two parameters (SDC and LS) are established, the design follows the accepted and widely used industry codes and standards.

Description of LMP/ASCE/SEI Integration Approach (proposed Option 2)

The LMP framework proposes a frequency-consequence (F-C) target to distinguish between unacceptable and acceptable event sequences. The F-C target is a frequency-versus-dose curve delineating ranges of acceptable risk for event sequences. The risk metric incorporates both the frequency of occurrence of the event sequences, including those termed licensing-basis events (LBEs),[5] and the associated radiological dose at the site boundary. The safety classification of SSCs in NEI 18-04 is made in the context of how the SSCs perform specific safety functions for each LBE sequence in which they participate. Risk-insights gained from the PRA model in identifying and selecting LBEs can be used to classify SSCs. The SSCs are classified as safety related (SR), non-safety related but safety significant (NSRSS), and non-safety significant (NSS). Safety-significant SSCs include all those SSCs classified as SR or NSRSS. The RIL 2021-04 gives an example of the LMP/ASCE/SEI Integration Approach. The

4 The IDP is a key element. As discussed in Appendix A and RIL 2021-04, there is no unique solution to meet a risk goal. It is important that the decisions being made account for the appropriate risk-informed considerations, such as DID and risk-balanced design.

5 LBEs are the event sequences considered in the licensing process to derive regulatory requirements. LBEs may include normal plant operational events; events anticipated to occur in the life of the facility; and off-normal events, including infrequent design-basis events.

safety classifications in the LMP framework are analogous to the proposed Part 53 Framework A categories of SR SSCs, NSRSS SSCs, and NSS SSCs.

The NRC staff developed the conceptual process in RIL 2021-04 to explain basic concepts and to explore the regulatory benefits of the RIPB/LMP seismic design framework. The design process builds on engineering approaches in structural or seismic engineering, maintains the familiar "deterministic" process for immediate use, and uses existing codes and standards to the maximum extent feasible. In RIL 2021-04, the SPRAs and the seismic design are interrelated (i.e., SPRAs are used to inform licensing decisions and aid the designer in assigning SSC design-performance target goals and design LSs). The concept of this iterative design process is to meet LMP criteria using combinations of variable seismic design requirements for individual SSCs and then examine their contributions to system-level performance using the SPRA. The proposed application of SPRA insights during the design process itself aims to arrive at a plant-level design that is both safe and more risk balanced, such that the seismic margins of individual SSCs are consistent with their risk significance within the overall system risk and performance goals of the plant. Design standards other than ASCE/SEI 43-19 and risk analysis methods other than SPRA may also be suitable to demonstrate compliance with risk goals, but this pre-decisional DG did not evaluate them.

Appendix A briefly describes this conceptual process, including an example, and RIL 2021-04 provides additional details. It is important to note that this approach determines the SDCs and LSs of SSCs that are used in the final design in accordance with ASCE/SEI 43-19 and related standards (see Section A.2 of Appendix A).

The pre-decisional draft regulatory positions and guidance are listed in Section C.

Additional Discussion of Proposed Option 3

This proposed option allows the use of a variety of performance and risk-analysis methods besides a traditional SPRA to demonstrate seismic safety.[6] Option 3 provides applicants with additional flexibility in developing approaches for evaluating seismic risks. These approaches depend on the size and type of commercial nuclear plant, the seismic hazard levels, the spectral shape of the seismic response spectra, and the specific safety measures being proposed to prevent and mitigate any potential adverse consequences that could result from a seismic event. Several alternatives could be considered in this proposed option, based on a broad spectrum of approaches that includes deterministic inputs; risk insights; and a combination of approximate, bounding, and conservative analyses and quantitative risk information. The current PRA standards recognize that the scope and attributes of a PRA can vary depending on the intended application and a graded PRA approach can apply.

An example is to initially assign SDCs and LSs for SSCs based on an understanding of potential risk or consequence considerations and other insights. The designer would design the SSCs using the ASCE/SEI suite of standards (see Section A.2 of Appendix A) and then confirm the design achieving targeted safety metrics through an appropriate risk-informed approach. The boxes in Figure 1 under proposed Option 3 reflect this overall approach.

The IDP discussed earlier is one of the mainstays in implementing the RIPB approaches. The IDP ensures that the process is acceptable and that the plant design meets the established criteria for adequate protection.

6 The draft regulatory position in Section C for this option proposes a requirement for an applicant to provide a risk-informed process with its basis. A pre-application engagement with the NRC staff could be very useful to obtain an early alignment on the process.

Relationship Between Proposed § 53.480 and Proposed Options 2 and 3

The section above titled "Proposed Option 1" explained how a seismic design using Option 1 satisfies the requirements of proposed § 53.480. The following paragraphs describe how proposed Options 2 and 3 could also be used to satisfy the requirements of § 53.480. The description is provided by using Option 1 as a reference since it is the established seismic design approach. Implementation considerations arise because proposed Options 2 and 3 expand the flexibility to allow the use of more than one DBGM in the seismic design, as allowed by § 53.480.

DBGMs: Multiple DBGMs replace the concept of a single SSE used in Option 1[7]. This set of DBGMs could include ground motions from the SDCs defined in ASCE/SEI 43-19 (e.g., SDC-3, SDC-4, and SDC-5), taking into consideration the safety functions, risk significance, and the required design performance of each SSC. These DBGMs should be derived from the site ground motion response spectra (GMRS) that may be developed following the guidance in RG 1.208. The RG 1.208 establishes a risk-informed and performance-based approach to define the site-specific earthquake ground motions for a site, which captures the regional and local seismic hazard. Apart from establishing minimum seismic design requirements, the SDC-5 DBGM that come from ASCE/SEI 43-19, is equivalent to the GMRS. Both the GMRS and the DBGM response spectra are derived from the same mathematical equations. This is also true for SDCs other than SDC-5 that are established using ASCE/SEI 43-19 equations with input from site-specific hazard analysis used for the development of a GMRS. In this way, the DBGM response spectra for all SDCs also captures the regional and local seismic hazard.

Minimum DBGM: The minimum earthquake design level(s) for proposed Options 2 and 3 should be designated based on the specific design of the SSCs and associated SDCs following provisions in ASCE/SEI 43-19.[8] This approach replaces the concept of minimum design requirement currently set at a peak ground acceleration (PGA) of 0.1g and a broad ground response spectrum. This change would be needed because multiple design response spectra (DRS)[9] are possible under proposed Options 2 and 3, and some of these may have a peak ground acceleration (PGA) lower than 0.1g. Therefore, the minimum earthquake design level(s) should be design specific (i.e., SDC specific).

Operating-Basis Earthquake Requirements: In the proposed Options 2 and 3, the operating-basis earthquake (OBE) ground motion is only associated with plant shutdown and inspections per § 53.480 (c)(2). This approach would remove the option of using an OBE-specific seismic design. This would be needed because the RIPB seismic design alternative could result in more than one DBGM level (i.e., SSCs can have different SDCs based on their risk significance). In addition, the RIPB seismic design includes the option to design using LSs other than a purely elastic design (LS-D). For some low SDCs and low hazard areas, the application of the current one-third of SSE ground motion criterion may result in a very low threshold criterion that may not exceed the empirical criterion of 0.2g and thus might not govern a decision to shut down.[10] However, this change would be necessary for high-hazard sites, which

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7 In 10 CFR 100.23, the nomenclature "safe shutdown earthquake ground motion (SSE)" is used to describe the DBGM. The notion of safe shutdown as a required safety function is central to LWRs. In the context of a broad spectrum of advanced reactor designs, the concept of safe shutdown as currently defined is not universally applicable.

8 ASCE/SEI 43-19 defines its own minimum PGA for various SDCs. For example, the minimum PGAs for SDC-4 and SDC-3 are 0.08g and 0.06g, respectively, with appropriate modifications to the entire DRS.

9 In ASCE/SEI 43-19, the DBGMs are described by DRS.

10 The current OBE shutdown criteria are described in American National Standard Institute (ANSI)/American Nuclear Standard (ANS) 2.23-2016, "Nuclear Power Plant Response to an Earthquake" (b 24).

may yield one-third of the DRS greater than 0.2g. The DRS check (i.e., the equivalent of SSE in the current approach) would be needed with a change to assure that a plant has not suffered permanent deformations that could lead to loss of safety functions, particularly when designed to LSs other than LS-D. The proposed changes accommodate more than one DBGM (and associated DRS).

Consideration of International Standards

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops safety requirements and safety guides for protecting people and the environment from harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports reflects an international perspective on what constitutes a high level of safety. To inform its development of this pre-decisional DG, the NRC staff considered IAEA safety requirements and safety guides pursuant to the Commission's "International Policy Statement," published in the *Federal Register* on July 10, 2014 (Ref. 14), and Management Directive and Handbook 6.6, "Regulatory Guides," dated May 2, 2016 (Ref. 15).

The IAEA has recently updated its Safety Standard for seismic design, SSG-67, "Seismic Design for Nuclear Installations" (Ref. 16). This IAEA guidance does not directly include RIPB design options like those described in this pre-decisional DG. However, these concepts are reflected in Section 7, "Seismic Margin To Be Achieved by the Design." For supporting new advanced reactor design and licensing, the IAEA staff is working on a technical document (TECDOC) called "Optimization of Protection against External Hazards."[11] This document presents a framework that uses an RIPB approach to consider site installation interactions.

C. STAFF PRE-DECISIONAL DRAFT REGULATORY GUIDANCE

This pre-decisional DG describes proposed RIPB seismic design approaches that the NRC staff considers potentially acceptable for meeting regulatory requirements of proposed Part 53 Framework A for the seismic design of advanced reactors. This pre-decisional DG proposes three options, as shown in Figure 1 and discussed in Section B.

Proposed Option 1 is an acceptable technical path to satisfy the proposed § 53.480 as it relates to the current seismic design approach (including use of SDC 5 and LS-D) and is not discussed further here since precedents are readily available and existing guidance (e.g., SRP, RG 1.233, etc.) could be adopted by Part 53 applicants. Therefore, the following proposed regulatory positions apply to Options 2 and 3. The discussions and proposed technical positions are framed in the context of two ASCE/SEI standards: ASCE/SEI 43-19 and ASCE/SEI 4-16. These standards are performance-based standards for nuclear facilities and contain or refer to associated material and construction codes and standards. These standards reflect the current state of practice for design and allow for establishing DBGMs and design LSs of SSCs considering risk information. Use of codes or standards other than the ASCE/SEI suite of standards could be considered with proper justification, provided they meet the intent of the proposed regulatory positions described here.

In Figure 1, proposed Option 2 is labeled as LMP/ASCE/SEI Integration Approach and proposed Option 3 is labeled as Graded SPRA and Other Risk-Informed Approaches. Appendix A describes a detailed process proposed for Option 2 and provides an example. The RIL 2021-04 contains a more

11 This information was obtained through an informal communication with an IAEA staff member.

detailed description with a discussion of considerations involved in implementing the process. Proposed Option 3 provides a broad spectrum of risk-informed choices (including graded PRAs) that would be suitable for design. The extent of explicit consideration of risk will affect the choices of suitable SDCs and LSs for individual SSCs.

[The staff plans to endorse ASCE consensus performance-based seismic design and analysis standards: ASCE/SEI 43-19 excluding Chapter 9 that the staff plans to endorsed in a separate pre-decisional DG and 4-16 excluding Chapter 12 that the staff plans to endorsed in a separate pre-decisional DG by reference with exceptions, additions, or clarifications provided by proposed regulatory positions C.3.5–C.3.9 contained in this section. Note that the excluded chapters from the endorsement of ASCE/SEI 43-19 and ASCE/SEI 4-16 describe technical provisions related to seismically isolated reactor design, construction and operational issues, which the staff plan to endorse in a pre-decisional DG titled "Seismically Isolated Nuclear Power Plants."]

C.1 Pre-decisional Draft Regulatory Positions for Proposed Option 2

C.1.1 Appendix A to this pre-decisional DG describes an acceptable process to determine SDCs and LSs for SR, NSRSS, and NSS SSCs in an SPRA. If an applicant uses an alternative process, it should explain the process and its technical basis.

C.1.2 This process uses the IDP considering both quantitative and qualitative risk information and other considerations listed in NEI 18-04 and RG 1.233. The applicant should describe the IDP and show that its attributes are in accordance with NEI 18-04 and RG 1.233.

C.1.3 Once the SDCs and LSs are established, the design of SSCs should be conducted based on ASCE/SEI 43-19, ASCE/SEI 4-16, and associated design codes, with the clarifications and exceptions noted in regulatory positions C.3.5–C.3.9. Alternative codes could be used with appropriate justification, provided they meet position C.1.4.

C.1.4 The final design is shown to meet the regulatory requirements, and an SPRA should be conducted in accordance with the guidance provided in RG 1.247.

C.2 Pre-decisional Draft Regulatory Positions for Proposed Option 3

C.2.1 The applicant should develop and provide an acceptable procedure and supporting technical basis to determine SDCs and LSs for SSCs. Alternatively, an applicant could assign SDCs and LSs to the SSCs based on their safety functions and risk importance using IDP considerations, provided these meet position C.2.4.

C.2.2 If the IDP in C.2.1 is used, considering both quantitative and qualitative risk information and other considerations listed in NEI 18-04 and RG 1.233, then the applicant should describe its IDP showing that necessary attributes are in accordance with NEI 18-04 and RG 1.233.

C.2.3 Once the SDCs and LSs are established, the applicant should design the SSCs based on ASCE/SEI 43-19, ASCE/SEI 4-16, and associated design codes referred therein, with the exceptions and clarifications noted in positions C.3.5–C.3.9. Alternative design codes could be used with appropriate justification, provided they meet position C.2.4.

C.2.4 The applicant should demonstrate that the final design meets the regulatory requirements and RG 1.233 through a graded SPRA (or other risk-informed approaches).

C.3 Pre-decisional Draft Regulatory Positions for Proposed Options 2 and 3

C.3.1 Minimum Design Ground Motion: The minimum earthquake design level(s) should be design- or SDC-specific. ASCE/SEI 43-19 defines its own minimum PGA for various SDCs. For example, ASCE/SEI 43-19 defines minimum PGA for SDC-4 and SDC-3 as 0.08g and 0.06g, respectively, with appropriate modifications to the entire DRS. The staff finds the minimum PGA levels and modifications to the DRS as outlined in ASCE/SEI 43-19 potentially acceptable.

C.3.2 Operating-Basis Earthquake: The OBE check uses both the certified seismic design response spectrum (CSDRS) and another spectrum, such as a site-specific design response spectrum (SSDRS), as applicable. The NRC interim staff guidance DC/COL-ISG-01, "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications," dated May 19, 2008 (Ref. 17), explains how to interpret the OBE exceedance in the context of both a CSDRS and an SSDRS. Conceptually, this interpretation is also applicable to multiple DRS that are possible for the alternative approaches of proposed Options 2 and 3. Considering these aspects, the following are the modified criteria from ANSI/ANS 2.23-2016, endorsed in RG 1.166, "Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake" (Ref. 18).

C.3.2.1 Response Spectrum Check: The OBE response spectrum check is performed using the lower of the following:

a. the DRS associated with the DBGM(s) used in the certified standard design, or

b. the DRS other than (a) used in the site-specific design of SSCs.

C.3.2.2 The OBE shall be considered to have been exceeded if both of the following occur:

a. Response spectrum check:

(i) the 5 percent damped acceleration response spectrum for any directional component (two horizontal and one vertical) of the earthquake motion at the site at frequencies between 2 and 10 hertz (Hz) exceeds the corresponding OBE response spectrum (one-third of a DRS) or 0.20g, whichever is greater, or

(ii) the corresponding OBE design spectral velocity or a spectral velocity of 6 inches per second, whichever is greater, is exceeded between 1 and 2 Hz.

b. Cumulative Absolute Velocity (CAV) check: The computed standardized CAV value from any component of the free-field earthquake record is greater than 0.16g-sec.[12]

C.3.2.3 The DRS[13] shall be considered to have been exceeded if both of the following occur:

12 Based on RG 1.166, this check does not apply if a free-field ground motion record is not available.

13 The DRS for this purpose is equivalent to the SSE in the ANSI/ANS 2.23 criteria.

a. Response spectrum check: The 5 percent damped acceleration response spectrum for any directional component (e.g., two horizontal and one vertical) of the earthquake motion at the site at frequencies between 2 and 10 Hz exceeds a DRS.

b. CAV check: The computed standardized CAV value from any component of the free-field earthquake record is greater than 0.16g-sec.

C.3.3 Radioactive Waste Management SSCs (Other Radiological Sources): RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" (Ref. 19), provides guidance for consideration of seismic loads. Seismic loads are prescribed as OBE or one-half SSE. In the alternative approach proposed in this pre-decisional DG, the DBGM can be associated with the SDC that will satisfy the dose criteria of NRC regulations.

C.3.4 Interactions among SR/NSRSS SSCs and NSS SSCs:[14] To be acceptable, each NSS subsystem should be demonstrated to be isolated from any seismic SR and NSRSS SSC by either a constraint or barrier or should be remotely located with regard to the SR and NSRSS SSCs. If this is not feasible or practical, then adjacent NSS subsystems should be analyzed according to the same seismic criteria as are applicable to the SR and NSRSS SSC. For NSS subsystems attached to seismic SR and NSRSS SSCs, the dynamic effects of the NSS subsystems should be simulated in the modeling of the seismic SR and NSRSS SSC. The attached NSS subsystems, up to the first anchor beyond the interface, should also be designed in such a manner that, during an earthquake ground motion of an applicable SDC, it will not cause a failure of the seismic SR and NSRSS SSC.

The following positions are based on a review of ASCE/SEI 4-16 and ASCE/SEI 43-19 by the NRC staff and a Brookhaven National Laboratory (BNL) review of ASCE/SEI 43-18 (draft) and ASCE/SEI 4-16. The BNL review findings are in RIL 2021-05, "Evaluation of ASCE/SEI 4-16 and ASCE/SEI 43-1-18 (Draft) for Use in the Risk-Informed, Performance-Based Seismic Design of Nuclear Power Plant Structures, Systems, and Components," issued July 2021 (Ref. 20).

C.3.5 Design Verification and Peer Review: This pre-decisional DG explains that applicants should conduct seismic design verification and peer reviews in accordance with ASCE/SEI 43-19 requirements in Section 10.1.1, "Seismic Design Verification," and Section 10.1.2, "Independent Seismic Peer Review." Peer reviews should focus on the concepts proposed in this pre-decisional DG that are not traditionally used and the alternate provisions in ASCE/SEI 43-19; for example, an alternate design in accordance with SDC grade (such as LSs C and B), nonlinear time domain analysis, deformation-based design, seismic separation and adjacent structure requirements, and nonlinear and three-dimensional approaches for building stability (e.g., sliding and overturning) analysis. The applicant should make design verification and peer review findings and reports available to the NRC.

C.3.6 Seismic Design Ground Motion: ASCE/SEI 4-16, Section 2.6, "Design Response Spectrum-Compatible Ground Motion Histories," and ASCE/SEI 43-19, Section 2.4, "Criteria for Developing Synthetic or Modified Recorded Time Histories," describe criteria for determining whether the generated time histories "match" or "envelop" the DRS. In particular, the ASCE/SEI guidance indicates that, if the response spectrum for a generated time history does not exceed the DRS by more than 30 percent at any frequency, a power spectral density (PSD) check is not

14 The applicable acceptance criteria for LWRs are in SRP Section 3.7.3, "Seismic Subsystem Analysis." The proposed position accounts for more than one seismic design classification of SSCs in the alternative approach.

needed. However, the ASCE/SEI DRS matching criteria alone may not prevent a power spectrum deficiency in the time histories, which could lead to underpredictions of in-structure response spectra (ISRS) and other frequency-sensitive structural responses.

This pre-decisional DG explains that a PSD check should be performed for multiple design time histories unless justification is provided that a stable mean ISRS can be achieved for the given number of design time histories. Therefore, either a single set or multiple sets of DRS matched, synthetic, or modified recorded time histories should be checked for power spectrum sufficiency by comparing them to a minimum target PSD function compatible with the DRS.

C.3.7 Missing Mass for Analysis of Structures: For mode-superposition time history analysis, ASCE/SEI 4-16, Section 4.2, "Linear Response-History Analysis Section," provides guidance on missing mass but also allows an alternative in which the number of modes included in the analysis should be sufficient to ensure that inclusion of the remaining modes does not result in more than a 10 percent increase in any response measure of interest. Similarly, for response spectrum analysis, ASCE/SEI 4-16, Section 4.3, "Linear Response-Spectrum Analysis Section," allows the 10 percent approximation approach, while providing an alternative to combine the high-frequency modes into a single residual mode.

In this pre-decisional DG, the 10-percent approximation is replaced with a calculation of the missing mass contribution to total response. Detailed information on the 10-percent approximation can be found in RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis" (Ref. 21), and NUREG/CR-6645, "Reevaluation of Regulatory Guidance on Modal Response Combination Methods for Seismic Response Spectrum Analysis," issued December 1999 (Ref. 22), which presents the technical basis for eliminating the 10-percent criterion. The missing mass contribution should be incorporated in both mode superposition time history analysis and response spectrum analysis to ensure accurate results.

C.3.8 Seismic Qualification by Analysis: ASCE/SEI 43-19, Section 8.2, "Seismic Qualification by Analysis," describes the seismic qualification by analysis using either an equivalent static analysis or a dynamic analysis method. For the equivalent static analysis of piping systems, Section 8.2.1.1 (Equivalent Static Analysis) of ASCE/SEI 43-19 indicates that the method described in paragraph N-1225, "Simplified Dynamic Analysis," of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, Division 1, Appendix N, "Dynamic Analysis Methods" (Ref. 23), is acceptable. For the dynamic analysis method, Section 8.2.1.2 (Dynamic Analysis) of ASCE/SEI 43-19 indicates that the approach in Appendix N to ASME Code, Section III, is to be used. However, the approach in Appendix N has not been generally reviewed and endorsed by the NRC, and in some cases, it presents very specific and prescriptive methods for safety-significant nuclear power plant piping systems. This pre-decisional DG explains that consistent with the goal of using an RIPB approach, flexibility is permitted in the analysis approach to meet the performance objectives of the SSCs. However, the use of methods other than those previously endorsed, such as those in Appendix N, should be peer reviewed. The applicant should make the peer review results and report available to the NRC.

C.3.9 Qualification by Testing and Experience Data: In ASCE/SEI 43-19, Section 8.3.2, "Demand for Qualification by Testing and Experience Data," the use of Institute of Electrical and Electronics Engineers (IEEE) 344, "IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations" (Ref. 24), for seismic qualification of electrical and mechanical components and the use of ASME/Qualification of Active Mechanical Equipment (QME)-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities" (Ref. 25), are generally acceptable. However, some aspects in testing and experience data require adequate justification to demonstrate their seismic adequacy. These aspects include restricting the frequency range of testing up to 33 Hz, the use of earthquake experience data, test experience data, and consideration of the effects due to high-frequency ground motions, if applicable (see RG 1.100, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants” (Ref. 26)). Therefore, this pre-decisional DG explains that the use of IEEE 344 and ASME QME-1 is acceptable; in addition, the qualification and justification should be consistent with RG 1.100.

D. IMPLEMENTATION

If the NRC staff should publish a DG on the topics discussed in this pre-decisional DG, then the NRC staff would explain in this section of the DG how the NRC would use the final RG in its regulatory processes. The NRC would also describe its use of the final RG in the context of the backfitting and issue finality provisions of preliminary proposed Part 53.

REFERENCES[15]

1. U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide (RG) 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Washington, DC.

2. NRC, trial RG 1.247, "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities," Washington, DC.

3. RG 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities"

4. NRC, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," Interim Staff Guidance, December 2021. (ADAMS Accession No. ML21336A702)

5. NRC, RG 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," Washington, DC.

6. Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Washington, DC, August 2019. (ADAMS Accession No. ML19241A472)

7. NRC, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," Commission Paper SECY-93-087, April 2, 1993.

8. NRC, DC/COL ISG 020, "Interim Staff Guidance on Implementation of a Seismic Margin Analysis for New Reactors Based on Probabilistic Risk Assessment," Washington, DC, March 2010.

9. NRC, Research Information Letter (RIL) 2021-04, "Feasibility Study on a Potential Consequence-Based Seismic Design Approach for Nuclear Facilities," April 2021, Washington, DC.

10. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 3.7.3, "Seismic Subsystem Analysis," Washington, DC.

15 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. The documents can also be viewed on line or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

11. American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) 43-19, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” New York, NY, 2020.[16]

12. American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) 4-16, “Seismic Analysis of Safety-Related Nuclear Structures,” New York, NY, 2017.

13. ASCE/SEI 7-10, “Minimum Design Loads for Buildings and Other Structures,” New York, NY, 2010.

14. NRC, “International Policy Statement,” *Federal Register*, Vol. 79, No. 132, p. 39415, Washington, DC, July 10, 2014. (ADAMS Accession No. ML14132A317)

15. NRC, Management Directive 6.6, “Regulatory Guides,” Washington, DC. (ADAMS Accession No. ML16083A122)

16. International Atomic Energy Agency (IAEA) Safety Standard, Specific Safety Guide SSG-67, “Seismic Design for Nuclear Installations,” Vienna, Austria, 2021.[17]

17. NRC, DC/COL-ISG-01, “Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications,” Washington, DC, May 19, 2008. (ADAMS Accession No. ML081400293)

18. NRC, RG 1.166, “Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake,” Washington, DC.

19. NRC, RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” Washington, DC.

20. NRC, RIL 2021-05, “Evaluation of ASCE 4-16 and ASCE 43-18 (Draft) for Use in the Risk-Informed, Performance-Based Seismic Design of Nuclear Power Plant Structures, Systems, and Components,” prepared with Brookhaven National Laboratory, Washington, DC, July 2021. (ADAMS Accession No. ML21194A062)

21. NRC, RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” Washington, DC.

22. NRC, NUREG/CR-6645, “Reevaluation of Regulatory Guidance on Modal Response Combination Methods for Seismic Response Spectrum Analysis,” Washington, DC, 1999.

[16] Copies may be purchased from the American Society for Civil Engineers (ASCE), 1801 Alexander Bell Drive, Reston, VA 20190 [phone: (800) 548-ASCE (2723)]. Purchase information is available through the ASCE Web site at http://www.pubs.asce.org.

[17] Copies of International Atomic Energy Agency (IAEA) documents may be obtained through its Web site: WWW.IAEA.Org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria.

23. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, Non-Mandatory Appendix N, "Dynamic Analysis Methods," New York, NY, 2004.

24. Institute of Electrical and Electronics Engineers (IEEE) 344-2013, "IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations," New York, NY, 2013. [18]

25. American Society of Mechanical Engineers (ASME)/Qualification of Active Mechanical Equipment (QME), ASME/QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," New York, NY, 2017.[19]

26. NRC, RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Washington, DC.

[18] Copies of IEEE documents may be purchased from the Institute of Electrical and Electronics Engineers Service Center, 445 Hoes Lane, P.O. Box 1331, Piscataway, NJ 08855 or through the IEEE's public Web site at http://www.ieee.org/publications_standards/index.html.

[19] Copies may be obtained from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016-5990. Telephone (212)591-8500; fax (212)591-8501; www.asme.org.

APPENDIX A

RIPB/LMP INTEGRATION APPROACH AND EXAMPLE— An Approach to Determine Seismic Design Categories and Design Limit States for Systems, Structures, and Components

To facilitate the discussion of regulatory concepts involved in the risk-informed, performance-based (RIPB) seismic design approach and to demonstrate its regulatory benefits, the U.S. Nuclear Regulatory Commission (NRC) developed the Licensing Modernization Project (LMP)/American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) Integration Approach and gave an example in its Research Information Letter (RIL) 2021-04, "Feasibility Study on a Potential Consequence-Based Seismic Design Approach for Nuclear Facilities," issued April 2021. This appendix gives an overview of the approach and an example of its application.

A.1 Overview of the LMP/ASCE/SEI Integration Approach

A.1.1 Purpose and Considerations

The purpose of the LMP/ASCE/SEI Integration Approach is to provide a framework to choose an appropriate combination of seismic design categories (SDCs) and limit states (LSs) for a structure, system, and component (SSC) considering its safety function and risk significance. This is a process that should occur before beginning the final design and defines SDCs and LSs for various SSCs for use in that design. Therefore, the rigor and accuracy of the information used in this approach should be of sufficient quality to allow for robust classification of SSCs into various SDCs. There are many considerations involved in choosing the appropriate SDCs and LSs. The RIL 2021-04 discusses these considerations and other factors involving consistency and coherency. Examples of such considerations include the targeted sites for a generic design, potential impacts related to shutdown and restart if an earthquake event occurs during operation, and potential future changes in perceptions of seismic hazards.

In addition, it is important to recognize that there are many SSCs in the various event sequences, and hence there are multiple possible performance combinations of the individual SSCs yielding a similar overall performance. Therefore, the decisions on selecting SDCs and LSs also involve factors other than quantitative risk goals, such as defense-in-depth (DID), ease of implementation of a design choice, and balance between prevention and mitigation. The integrated decision process described in Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," issued August 2019, is crucial from this perspective and is embedded in the approach described here.

A.1.2 Overview of LMP/ASCE/SEI Integration Approach

Figure A-1 illustrates the LMP/ASCE/SEI Integration Approach. This procedure was developed in the context of ASCE/SEI 43-19, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," and associated codes. However, the seven-step approach is flexible enough to allow the use of seismic codes and standards other than ASCE/SEI 43-19.

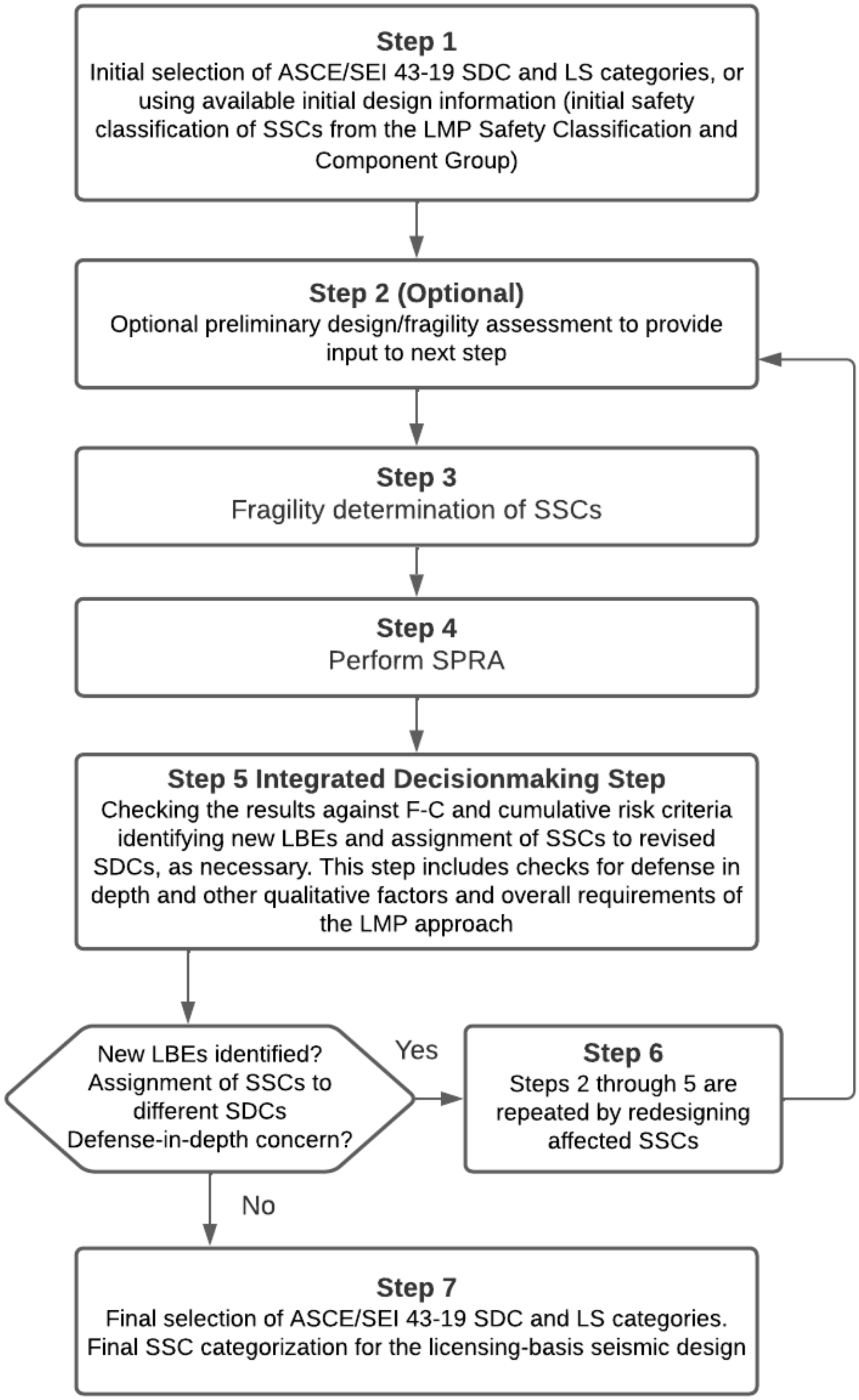

Figure A-1. LMP/ASCE/SEI Integration Approach

<u>Step 1</u>: Initial Selection of the ASCE/SEI 43-19 SDC and LS Categories

This step is related to the initial selection of SDC and LS for SSCs that are modeled or will be modeled in a seismic probability risk assessment (SPRA). For example, for advanced reactors whose designs are already in progress or have been completed and whose seismic design is based on an approach akin to those in ASCE/SEI 4-16, "Seismic Analysis of Safety-Related Nuclear Structures," and ASCE/SEI 43-19, using SDC-5 and LS-D requirements, initial fragilities needed in Step 3 can be assigned based on currently available generic fragility values. These initial values are used to help determine whether a different SDC and LS can be assigned through this iterative process.

For newer advanced reactor designs, there are more seismic design options using combinations of SDCs and LSs, such as SDC-5 and LS-D, SDC-5 and LS-C, SDC-4 and LS-D, which could be selected at the onset.

Additional considerations on the initial selection of SDC and LS categories include regulatory requirements, stability of the designs, ease of design, and available information. The selection of LS for an SSC is related to its intended safety function.

<u>Step 2</u>: Preliminary Design/Fragility Assessment (optional)

Conduct a preliminary design and a fragility assessment to determine whether more realistic fragilities of important SSCs are needed. This optional step may only be necessary in subsequent iterations to implement Step 5 more comprehensively. In most cases, once the SDC and LS categories are chosen, one can proceed to Step 3 to assign fragilities. However, in some cases, it may be necessary to have a better understanding of the design or a better estimate of a fragility to support a more informed and robust decision. This step provides such an opportunity.

<u>Step 3</u>: Fragility Determination

Based on the assignments of SDC and LS categories in Step 1 and based on available data**,** fragilities can be calculated or can be estimated using generic information, engineering judgment**,** or experimental data. For the purposes of determining SDC and LS categories, precise values of fragilities are not necessary, provided they are within a realistic range. The current generic database reflects the current practice of designing safety-related SSCs using SDC-5 and LS-D categories. The RIL 2021-04 discusses some of the considerations involved in adjusting these fragilities for different SDC and LS combinations.

<u>Step 4</u>: Perform Seismic PRA

This step involves performing an SPRA using the fragility values of Step 3. The LMP approach requires an SPRA at the stage of design development. The RIL 2021-04 describes some options for choosing probabilistic seismic hazard(s) for this stage. Output from the SPRA includes frequency-consequences (F-Cs) for event sequences, dominant contributors, and ranking of sequences.

<u>Step 5:</u> Integrated Decision-making

In this step, the integrated decision-making team checks the proposed classification against the risk, DID, and other criteria. The team evaluates the results of the initial probabilistic risk assessment (PRA) to determine whether the individual event sequence risks are within the F-C target goals and whether the integrated risk criteria are met. This team also evaluates DID adequacy, reliability, and other qualitative factors related to risk-informed decision-making (e.g., balance between prevention and mitigation, and avoidance of singleton failures that control the risk), and other LMP guidelines. The LMP safety classification and component group may identify opportunities to design SSCs to a less stringent SDC or LS. This feedback is provided to the seismic design and SPRA teams to recalculate the SSC fragilities, as needed.

Step 6: Iteration

Steps 2 through 5 are repeated to optimize the design to meet all regulatory requirements and the desired safety and cost goals. This process ends when final selections of SDCs and LSs are made for SSCs.

Step 7: Final SSC Categorization for Seismic Design

Based on the iterative process described in Step 6, a determination is made for the final SSC categorization to support of the seismic design. It becomes the basis for the final seismic design and licensing of a certified design for a plant. This categorization and associated fragilities will be used in the final SPRA.

A.2 Seismic Design Process (Applicable to Both Proposed Options 2 and 3)

This section is included to provide a better understanding of how the outcome from the above approach should be used in the actual design. As shown in Figure A-1, the outcome of the LMP/ASCE/SEI Integration Approach is a designation of SDC and LS categories for each SSC or groups of SSCs. The following steps provide an overview of how the design process may proceed based on current practice and consistent with ASCE/SEI 43-19 and ASCE/SEI 4-16, with exceptions, additions, and clarifications noted in this Appendix:

(1) Design response spectra for each SDC are derived from the probabilistic seismic hazard analysis results using the selected SDC and the procedure in ASCE/SEI 43-19.

(2) A seismic response analysis is conducted using ASCE/SEI 4-16 methods—similar to current requirements.

(3) The design of SSCs follows engineering approaches in appropriate codes and standards referenced in regulatory guides and sections of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."

(4) The design of building elements and anchorages is created to meet American Concrete Institute (ACI) 349, "Code Requirements for Nuclear Safety-Related Concrete Structures," and ACI 359, "Code for Concrete Reactor Vessels and Containments," and American National Standards Institute (ANSI) American Institute for Steel Construction (AISC) N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities."

(5) The design of mechanical equipment; piping systems; cable tray systems; and heat, ventilation, and air conditioning systems follow the American Society of Mechanical Engineers (ASME) codes.

(6) Seismic design and qualification of electrical components follow Institute of Electrical and Electronics Engineers (IEEE) 344-2013, "IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations."

(7) Design alternatives (e.g., base isolation) and sophistication (e.g., nonlinear analysis) are pursued, as appropriate.

The primary difference in the LMP/ASCE/SEI Integration Approach compared to current design practice is the possibility of selecting different combinations of SDC/LS for various SSCs informed by their contributions to system- and plant-level safety, as opposed to assigning SDC-5 and LS-D to all safety-related SSCs.

In the LMP framework, risk evaluations are performed through the SPRA. Under current requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," the final SPRAs are conducted at the following three completeness stages, with each SPRA relying on the available design information at that stage:

(1) certified design application
(2) combined license application considering site-specific hazard, site, and other information
(3) before fuel loading, considering as-designed, as-built, and other operating conditions

Plant- and site-specific fragility analyses and SPRAs will then follow the accepted methodologies specified in the ASME non-light-water reactor (non-LWR) PRA standard and trial RG 1.247, "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities."

A.3 Example Application

A.3.1 <u>Introduction and Objective</u>

In this example, a simplified SPRA model of an advanced reactor design is used to demonstrate basic concepts of the LMP/ASCE/SEI Integration Approach. The primary objective of this example is to show the impacts on individual event sequence frequencies and consequences from alternative selections of ASCE/SEI 43-19 SDC and LS categories for design. These results will show at a conceptual level whether alternative designs are feasible, how the event sequence frequencies vary, and how the approach is applied. This simple example does not explore the following topics (although RIL 2021-04 discusses these in more detail):

- effect on cumulative risk from all internal and external hazards
- changes in risk insights, such as changes in dominant sequences, dominant contributors, and non-seismic failures
- complex decision and implementation considerations
- impact of other regulatory and technical considerations

A.3.2 <u>Description of Example</u>

Figure A-2 shows the event tree model used in this example. This is a simplified event tree from a publicly available model for an advanced non-LWR reactor. However, hazard and fragilities used in this demonstration have no relationship to the earlier analysis. Therefore, this is a hypothetical model. This simplified model still maintains the functional coherency of the event sequences, so that the dose consequences from the earlier analysis can be used.

Simplification is in terms of deleting some top events and generally representing the failure of a top event through a single component failure. An event sequence related to “Building Failure” was added to evaluate the impact on event sequence frequencies with alternative SDCs and LSs. No dose consequences are calculated for these event sequences. Two event sequences, Sequences 3 and 6, lead to dose consequences labeled Dose 1 (lower dose) and Dose 2 (medium dose) in Figure A-2., Sequence 7 (building failure) is assumed to lead to some dose consequences, but doses are not estimated.

Table A **-** 1 lists the four components whose fragilities are used in the simple SPRA model, where the heat transport system is denoted as HTS and the shutdown cooling system is denoted as SCS. A_m represents median capacity of a component and $ß_c$ the composite uncertainty.

Table A-1. Fragilities for Different Design Cases

	LMP Design 1 SDC-5/LS-D Base Case		**LMP Design 2 SDC-4/LS-D**		**LMP Design 3 SDC-5/LS-C**		**LMP Design 3 SDC-5/LS-C Sensitivity S1**	
	A_m	β_C	A_m	β_C	A_m	β_C	A_m	β_C
Shear Wall	2.9	0.46	1.45	0.46	1.93	0.46	1.93	0.46
Primary Boundary	2.9	0.46	1.45	0.46	1.93	0.46	1.93	0.46
HTS Cooling	1.24	0.40	0.62	0.4	1.24	0.4	0.93	0.4
SCS Cooling	1.24	0.40	0.62	0.4	1.24	0.4	0.93	0.4

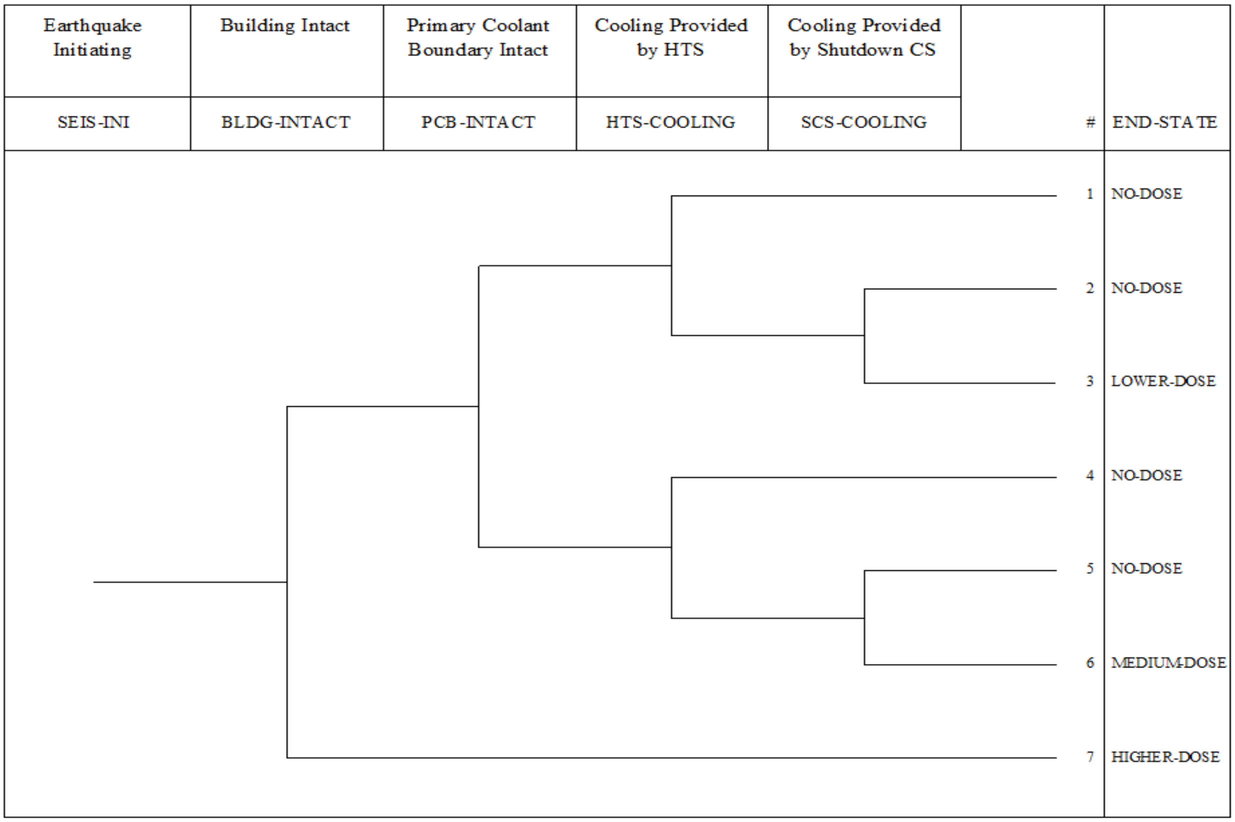

Figure A-2. Simplified event tree for a hypothetical advanced reactor

A.3.3 Analysis Approach and Results

The analysis approach is demonstrated in the context of the stepwise procedure described in Section A.1.2.

Step 1: This step uses the SPRA model of Figure A-2, based on the available design information at the time. For the purposes of this demonstration, the following assumptions are made:

(1) This design is assumed to be a generic design for a site in the central or eastern United States (CEUS).

(2) The design input is assumed to be similar to that being used for recent advanced light-water reactor designs. Specifically, the design input corresponds to SDC-5.

(3) The initial design uses the LS-D, similar to the current practice. Thus, the initial design is in accordance with SDC-5 and LS-D. This is considered the base case, LMP Design 1 in Table A - 1.

Step 2 (optional): For this demonstration example, no additional specific analysis is carried out in the initial and subsequent iterations.

Step 3: Because the design is assumed to reflect the SDC-5 and LS-D design, the initial trial can use the generic fragility values reflecting plant design in the CEUS. RIL 2021-04 describes additional considerations involved in choosing the fragility values, along with examples of some generic approaches. That report also discusses how to adjust the fragilities to reflect different design alternative combinations (e.g., SDC-5 and LS-C, SDC-4 and LS-D).

Table A-1 lists the initially assigned fragility values, in terms of median capacity, $A_{m,}$ and composite uncertainty, $ß_c$, values for LMP Design 1, the base case.

Step 4: For the purposes of performing an SPRA, two CEUS sites are chosen; Site A, reflecting a relatively high-hazard site in the CEUS, and Site I, reflecting a more moderate seismic hazard site. Figure A - -3 shows the two mean seismic hazard curves that are used for quantification. These curves are based on results submitted to the NRC in response to a generic request in the aftermath of the 2011 Fukushima Daiichi incident.

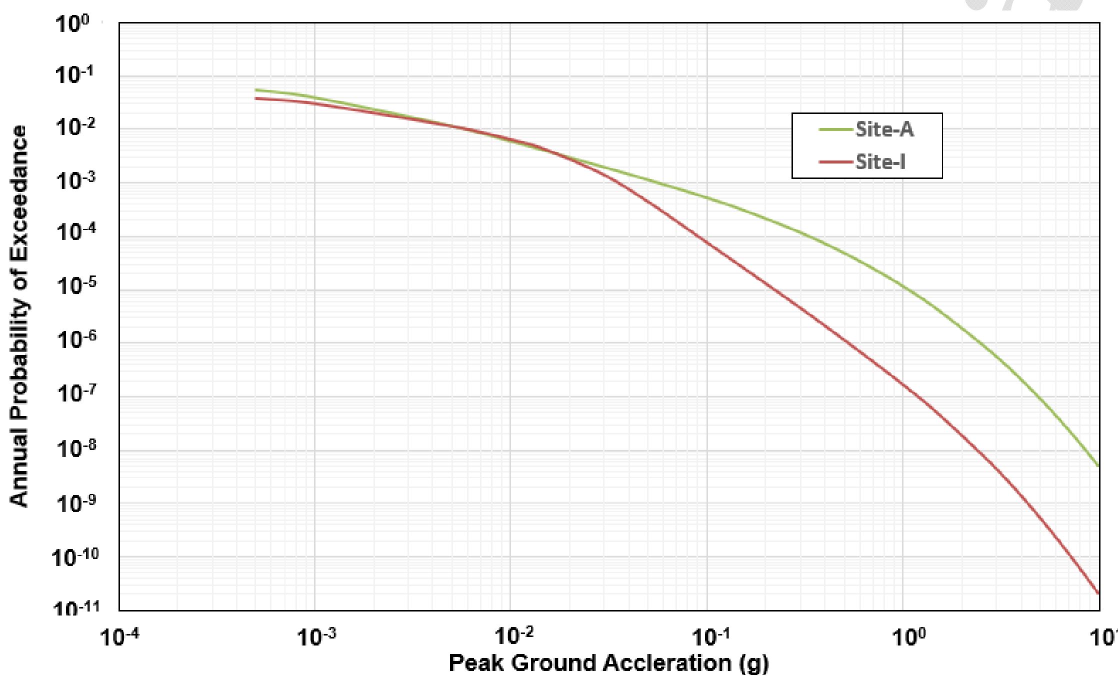

Figure A-3. Hazard curves for Sites A and I

The red squares in Figure A-4 show the results of quantification by using the hazard curve for Site A for the LMP Design 1 for Sequences 3 and 6. Doses for Sequences 3 and 6 are the same as in the original model as, phenomenologically, these sequences are the same. Figure A-4 also shows the simplified version of the target F-C curve of the LMP framework. The frequency of Sequence 7 is in order of 10^{-6} based on assumed fragility.

Figure A-5 shows the same results for Site I. Frequencies of event sequences that result in doses are significantly lower for the LMP Design 1 compared to Site A.

Step 5: This is the integrated decision-making making step. By looking at the results of the computed frequencies and doses for two sequences and comparing them to the F-C target for both sites, there is significant margin, even considering potential future hazard changes and other factors, as discussed in RIL 2021-04. Therefore, it was decided that two additional design cases should be evaluated by further iterating. The two cases are designated as LMP Design 2 that uses

SDC-4 and LS-D design, and LMP Design 3 that uses SDC-5 and LS-C, respectively, as shown in Table A-1. All the components were assumed to be designed to new categories.

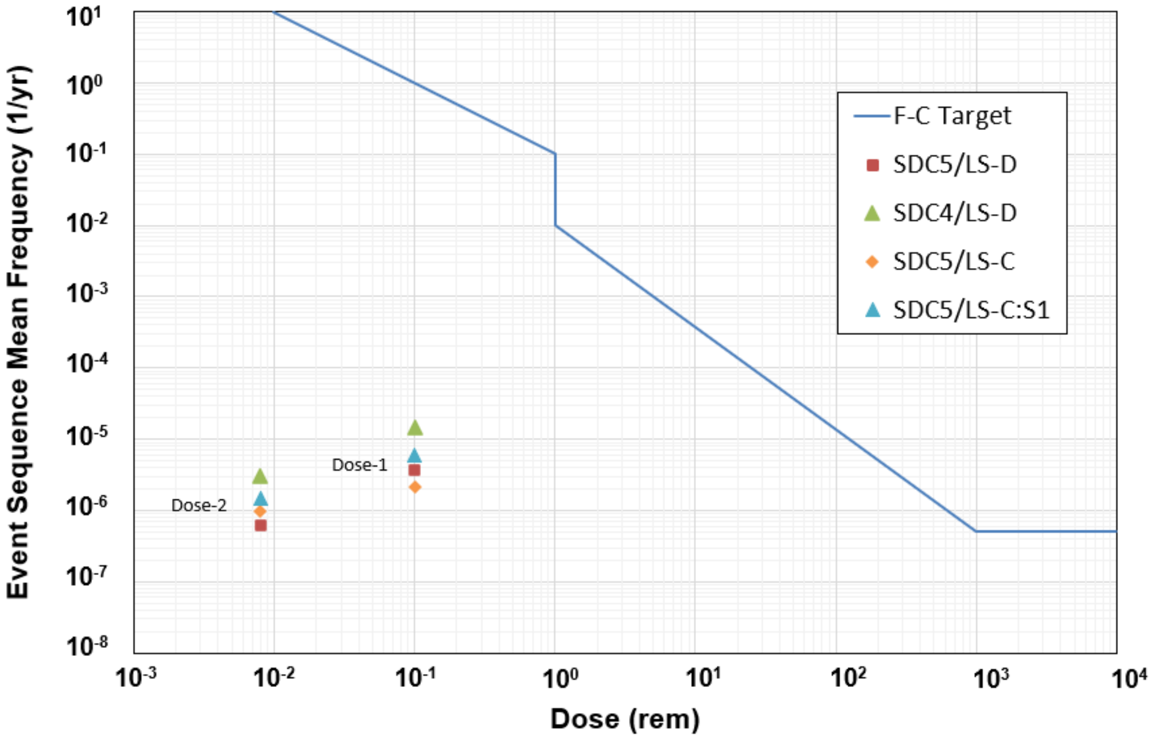

Figure A-4. Results of quantification for Site A

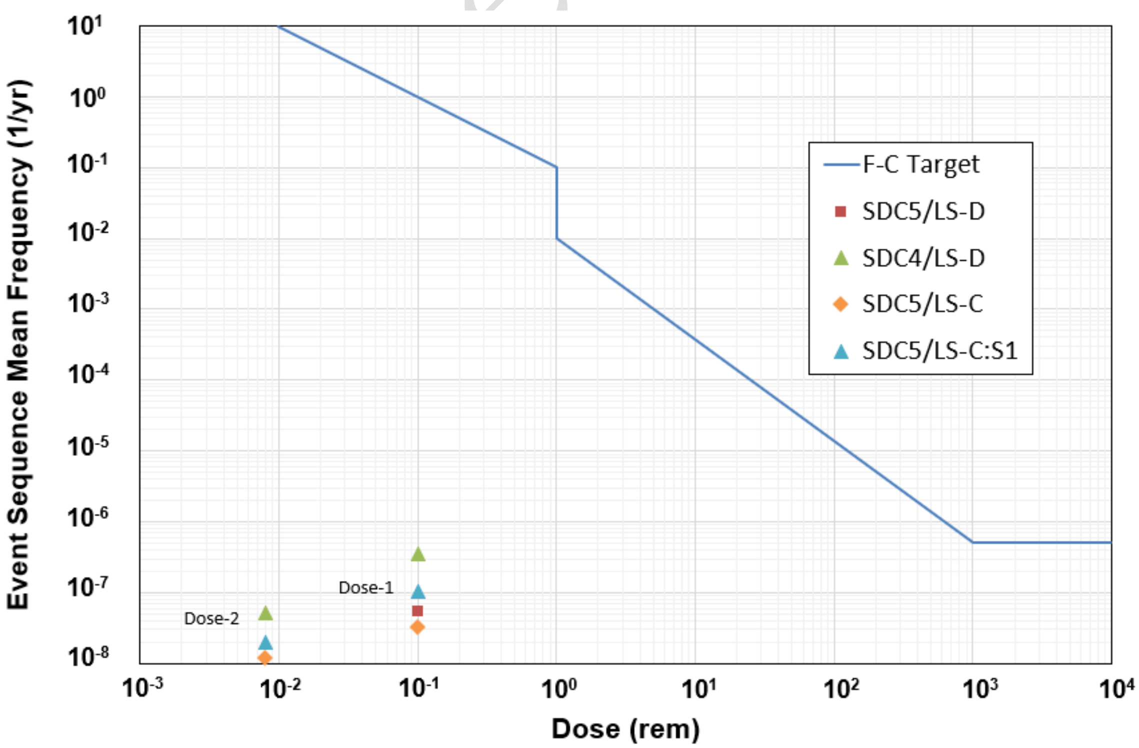

Figure A-5. Results of quantification for Site I

Step 6: This is the step that returns to Step 2 if design changes are being considered. In this demonstration example, the fragilities for additional design cases, LMP Design 2 and LMP Design 3, were adjusted based on the procedure discussed in RIL 2021-04 and certain assumptions. For example, the DBGM for SDC-4 is about half of the SDC-5 DBGM. Thus, for a linear system, under certain circumstances, the median seismic capacity of a component designed to SDC-4 would be half of the median capacity if that component were designed to SDC-5. Table A-1 shows the fragilities for LMP Design 2 and Design 3, reflecting such adjustments. The results of quantifications for these two additional design cases are shown in Figures A-4 and A-5 for Site A and Site I, respectively, with designations SDC-4/LS-D (LMP Design 2) and SDC-5/LS-C (LMP Design 3).

Step 7: This is the final step in which decisions are made with respect to classification of SDCs and LSs for various SSCs. Looking at the quantification results in Figures A-4 and A-5, for this hypothetical design with low-dose consequences, an adequate design may be acceptable with SDC-4 and LS-D.

A.4 Observations and Conclusions

The above example is highly simplified to demonstrate the basic concepts of the RIPB seismic design approach. A full SPRA model needs to be used in actual applications to fully reflect different outcomes in the frequency-dose calculations. As the fragilities are changed to reflect alternate seismic designs, changes in the dominant sequences and contributors are likely. The cumulative risk also needs to be considered from the seismic and other initiators.

This simplified example does demonstrate the feasibility of this approach and opportunities to modify the design considering safety and cost benefits.