A PROPOSED ALTERNATIVE RISK-INFORMED AND PERFORMANCE-BASED REGULATORY FRAMEWORK FOR SEISMIC SAFETY AT NRC REGULATED FACILITIES—TASK 3

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ABSTRACT

The commercial nuclear power plant industry has initiated the Licensing Modernization Project (LMP) to enhance the risk-informed and performance-based (RIPB) regulatory basis for licensing and regulating the safety of advanced nuclear power reactors. The LMP framework is supported by both the U.S. Department of Energy (DOE) and the U.S. Nuclear Energy Institute (NEI). The LMP framework relies heavily on RIPB concepts and approaches that together integrate the defense-in-depth philosophy. The U.S. Nuclear Regulatory Commission (NRC) is considering adopting some or all the LMP proposals and issued a Draft Regulatory Guide (DG) DG-1353 for public comment.

Although the LMP framework contains significant details about how its concepts should work for many aspects of reactor safety including external hazards, it does not yet explicitly address how to incorporate seismic performance criteria into the physical design of structures, systems and components (SSCs). This is particularly true for advanced reactor designs that rely on passive safety controls, have significantly different facility footprints, or utilize accident-tolerant fuels.

The objective of the project to which this report belongs is to evaluate if and how to implement recent improvements to existing industry codes and standards in nuclear seismic design with the LMP framework. The project also seeks to identify any implementation and regulatory hurdles that could inhibit broad application of these improvements in codes and standards within the LMP concepts. This report documents initial efforts to fulfill the stated objective. Preliminary conclusions are as follows:

- An initial approach has been developed that aligns the LMP concepts with the American Society of Civil Engineers (ASCE) standard ASCE 43 for seismic design. In this approach the seismic design is integrated with the seismic probabilistic risk assessment (SPRA) and is referred to as LMP/ASCE 43 Integration Approach. The general process can be used for both Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 and Part 52 applications. The process is technology inclusive and applicable to array of advanced nuclear power reactor designs.

- There are no obvious impediments in implementing the enhanced risk-informed seismic design requirements based on the LMP concepts and recent updates to the several seismic design standards and regulatory guides. Several technical, programmatic, and regulatory considerations for the successful implementation of the enhanced RIPB seismic concepts were identified.

- The major benefits of the proposed RIBP enhancements come from the flexibility to assign different seismic design categories (SDCs) (thus, different design basis ground motion levels) and different design performance limits, that is, different limit states (LSs), to SSCs considering their risk-significance and other risk-informed decision-making factors (in contrast to the current approach that uses a single design basis earthquake and a very stringent elastic LS for all SSCs, irrespective of their risk significance). Thus, in this enhanced RIPB seismic design process, the safety margins of individual SSCs are controlled according to their contribution to system-level and plant-level risk, thereby reducing unnecessary conservatisms (or providing additional margins, as needed) and achieving a more risk balanced design.

- Stakeholder interactions will be crucial for broader acceptance and refinement of the proposed process and for developing additional cooperative activities.

- Additional analyses are recommended to fully demonstrate the implementation and to identify associated pros and cons of the proposed changes to the seismic design process within the larger context of seismic design, seismic safety, optimized costs, and existing seismic safety regulations.

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EXECUTIVE SUMMARY

Over the past few years, the United States (U.S.) commercial nuclear power industry initiated the licensing modernization project (LMP) to enhance the risk-informed and performance-based (RIPB) regulatory basis for licensing and regulating the safety of advanced nuclear power reactors. The LMP framework[1] includes appropriate risk targets [both for overall plant-wide risk and for event sequences with multiple structures, systems, and components (SSCs)] and an approach to classifying safety-important 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 assuring defense-in-depth. Finally, it more directly applies probabilistic risk assessment (PRA) modeling as a basis for much of the understanding that would support safety decision-making (e.g., NEI, 2018; INL, 2018). The LMP framework is supported by both the U.S. Department of Energy (DOE) and the U.S. Nuclear Energy Institute (NEI). The LMP framework relies heavily on RIPB concepts and approaches that together integrate the defense-in-depth philosophy. For its part, the U.S. Nuclear Regulatory Commission (NRC) is considering adopting some or all the LMP proposals and issued a draft Regulatory Guide (RG) RG-1353 (NRC, 2019), for public comment as part of the staff's deliberation process.

While the LMP framework considers external hazards in terms of functional design bases, it does not yet explicitly address how to incorporate seismic performance criteria into the physical design of SSCs. This is particularly true for advanced reactor designs that rely on passive safety features, have significantly different facility footprints, or utilize accident-tolerant fuels. Although the development of RIPB approaches for seismic design and seismic safety analyses date back to initiatives started in the 1970s[2], these RIPB approaches continue to evolve. This ongoing evolution of RIPB thinking is evident by updates to the many standards and regulatory guides that apply to seismic hazard and seismic design, such as NRC RG 1.208 (NRC, 2007a), DOE Standard 1020-2016 (DOE, 2016), and American Society of Civil Engineers (ASCE) Standards 4 (ASCE, 1998) and ASCE 43[3] (ASCE/SEI, 2005), among others.

Although many applications of seismic probabilistic risk assessment (SPRA) over the years have confirmed that there are significant margins in existing seismic design in-terms of ability to perform required safety functions[4], some of the fundamental issues associated with the deterministic design procedures embedded in NRC's regulatory approach remain. These issues relate to not explicitly considering the behavior of safety-significant SSCs for beyond design basis ground motions and to uneven margins that are inconsistent in terms of their contribution to plant-level safety or risk significance. In addition, the compartmentalized approach to design (e.g., considering each SSC separately) results in excessive conservatisms in many cases that may not contribute to overall plant safety and that do not provide a consistent way to achieve the desired seismic capability of the overall plant. SPRA reflects information on the seismic capability of a plant considering aspects of design, construction, and operations. The increased

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[1]Framework in this context is meant as a conceptual structure to support development of the practical components needed to construct and regulate Advanced Light Water Reactors (ALWRs) and advance non-light water reactors.
[2]The development of probabilistic seismic hazard analysis began in the 1960s with a seminal publication by Prof. C. Allin Cornell (Cornell, 1968).
[3]The currently published ASCE Standard 43 is Version 43-05, published in 2005. A new version is planned to be published soon. The basic philosophy of both versions is same. For the purposes of this Executive Summary, ASCE 43 is cited without the year of publication.
[4]In new license applications under 10 CFR Part 52, SPRAs are used to demonstrate the overall design margin considering beyond design basis ground motions.

use of SPRAs, in general, the current movement toward a more RIPB licensing framework, and recent extensive experience with SPRAs provide a unique opportunity to incorporate RIPB concepts in updates to NRC regulations and guidance that govern seismic design; especially in ways that lead to a more uniform, balanced, and safety- and risk-consistent plant.

Objectives

The NRC staff within the Office of Nuclear Regulatory Research (RES) initiated the project governing the work described in this report to identify and evaluate potential improvements to the existing regulatory basis for seismic design. The revised and enhanced RIPB framework will be offered as an alternative to the more deterministic approach currently in use. The objective of this project is to evaluate if and how to implement the enhanced RIPB concepts of the LMP framework together with recent improvements to industry codes and standards in nuclear seismic design. This project also seeks to identify any implementation and regulatory hurdles that could inhibit broad adoption and application of the LMP framework.

Specifically, the NRC staff seeks to develop a set of preliminary technical and regulatory insights that support (i) integration of the LMP concepts within the current NRC regulatory philosophy and regulations; (ii) adoption of appropriate risk metrics; (iii) understanding system-level and plant-level performance behavior relative to the performance of individually designed components; (iv) integration of seismic-induced failures with non-seismic failures and human errors; (v) quantification of the relative risks from earthquakes compared to those from other accident initiators; and (vi) consideration of other desirable factors for overall plant safety, such as defense-in-depth and risk-balanced profiles (i.e., balance between prevention and mitigation, more uniform margins, avoidance of singleton failures that may control risk, etc.).

Those insights are intended to be technology-inclusive and explicitly incorporate RIPB evaluation techniques that can be used to license future commercial advanced nuclear reactors. Any new regulations or guidance for seismic design stemming from this project should be integrated within the broader NRC RIPB framework and build on existing risk-informed approaches in structural and seismic engineering. Finally, the recommended insights should be rooted within existing practices (e.g., using existing codes and standards to the maximum extent practicable) such that they can be implemented within the current NRC regulations [e.g., Title 10 of the *Code of Federal Regulations* (10 CFR) Parts 52 and Part 50] or with reasonable and practical updates to these regulations and associated guidance.

The current work described in this report examines ideas and processes to align the LMP concepts within the current RIPB framework for seismic design and safety and identifies future potential activities to support the NRC’s objectives. Future activities are identified to both demonstrate the feasibility and validity of the process developed so far through some simple SPRA models and to obtain and include stakeholder input through public workshops. Depending on the outcomes of the workshops, a cooperative effort with the industry may be considered as a way to conduct more detailed evaluations of the concepts and process through the use of detailed SPRA models. These more detailed evaluations will provide additional technical and regulatory insights into implementation issues and will further clarify the concepts and process described in this initial report.

Overall Approach

An initial approach has been developed that aligns the LMP concepts with the American Society of Civil Engineers (ASCE) standard ASCE 43 for seismic design. This process is referred to as

LMP/ASCE 43 Integration Approach. The underlying strategy used to achieve the stated objectives considered the performance of individual SSCs as well as the role they play in an event sequence, in contrast to current regulations, in which every individual safety-related SSC is designed to the same seismic criteria [e.g., the Safe Shutdown Earthquake (SSE)[5]] irrespective of the role the SSC plays in the overall system performance. This new strategy embeds risk targets into the criteria used to determine the acceptability of the seismic design of each SSC, and the way these risk targets "work" requires explicit consideration of the behavior of each important SSC when seismic demands exceed the design basis ground motion. This new philosophy, which explicitly uses the SPRA and considerations such as defense-in-depth and safety margins to inform the engineering design of the plant, as well as the design of each individual SSC, to be flexibly tailored to the desired performance in the relevant event sequences. In doing so, the strategy achieves desired safety goals and allows nuclear plant designers and operators greater flexibility in how the overall seismic design can meet system-level acceptability criteria, as well as plant-level acceptability criteria. It also further enables allocating resources where they matter the most for safety.

The LMP framework proposes a Frequency-Consequence (F-C) Target to identify acceptable accident event sequences. The F-C Target is a frequency-versus-dose curve delineating ranges of acceptable risk for event sequences. The risk metric incorporates the frequency of occurrence of the event sequences, termed licensing basis event (LBE)[6], and the associated radiological dose to the public at the site boundary. If the F-C metric is not met, the design is modified, and the process is iterated. In addition to recommending that acceptable individual event sequences lie below the F-C Target, the LMP framework accounts for aggregate risk by adding the product of the frequency and the dose of the event sequence over all sequences and comparing this total risk with cumulative risk targets. If the cumulative or integrated criterion is not met, further enhancements to the facility design may be necessary to comply with the total risk criterion.

It is important to note that under the LMP framework, the safety classification of SSCs already accounts for special treatment, such as differences in maintenance and operating requirements. However, it does not yet explicitly address how to incorporate seismic performance criteria into the physical design of an SSC. Thus, the LMP framework represents a change in the governing design philosophy, in which designing an individual SSC for seismic safety is tied to the SSC's role in the overall safety, as "measured" by the SSC's contribution to the risk of those event sequences in which it participates. Table 1 describes key conceptual differences between the current approach for seismic design of nuclear power plants and the proposed approach based on the LMP and enhanced RIPB concepts in ASCE 43.

Table 1 also highlights important benefits of using enhanced RIPB seismic design concepts. These enhanced RIPB concepts allow flexibility to assign a different seismic design category (SDC) to individual SSCs (i.e., flexibility in selecting the different design basis ground motion levels), and by permitting selection of alternate design limit states (LS), the desired margins can be maintained consistent with the SSC's contribution to overall plant risk. These concepts avoid

[5]SSE is the design basis ground motion that is used to determine seismic loads for individual SSCs, and these loads, along with other appropriate load combinations, are used for the design. The nuclear power plant must be designed such that, if the SSE Ground Motion occurs, critical SSCs will remain functional and within applicable stress, strain, and deformation limits.

[6]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 (DBEs)

excess conservatisms that do not provide commensurate safety benefits (or identifies situations where additional margins maybe needed). These concepts are practicable because once the SDC and LS are chosen, the design processes are essentially the same as those used currently[7].

Table 1. Comparison of the current seismic design and proposed alternative approach

Current Seismic Design Approach	ASCE 43/LMP Approach
Safety Classification: All safety related systems, structures, and components (SSCs) for the seismic design purposes are considered Seismic Category 1 SSCs.	Safety Classification: Alternate safety classification is a part of the licensing modernization project (LMP) framework considering risk significance. American Society of Civil Engineers (ASCE) 43 allows possibility of alternate seismic design categories (SDCs)[8] considering the desired level of design performance and consistent with the risk significance.
Design Basis Ground Motion: All SSC-1 SSCs are designed to one ground motion level corresponding to the Safe Shutdown Earthquake (SSE) or design basis ground motion. The current site-specific design basis ground motion corresponds to the highest level determined using the Regulatory Guide 1.208 approach and it is based on the hazard exceedance frequency for a performance goal of 1×10^{-5} per year.	Design Basis Ground Motion: Design ground motions for each SDC are derived considering the performance target and margins associated with the design process. Thus, there is no single design basis ground motion for all SDC categories. The ground motions are based on hazard frequencies for targeted performance goals that vary with each SDC.
Design Performance Criteria: No explicit numerical criteria are defined. The design limits are associated with elastic behavior resulting in significant safety margins beyond the design basis ground motion.	Design Performance Criteria: A quantitative design performance criterion is associated with each SDC. Alternate design limit states (LS) (e.g., allowing inelastic behavior) are permitted, considering the desired design performance and margins.
Design Procedures: The design is carried out using deterministic seismic response analyses for the SSE ground motion to establish the seismic demand. The physical design of an SSC is accomplished using established construction and engineering standards, such as those published by the American Concrete Institute (ACI), American Institute of Steel Construction (AISC), American Society of Mechanical Engineers (ASME) and others.	Design Procedures: The design employs seismic response analyses based on the ground motions that correspond to the assigned SDC categories. Seismic demands are adjusted based on the selected LS. The physical design of an SSC is accomplished using established construction and engineering standards, such as ACI, AISC, ASME and others.

The following activities were conducted to support the project objectives:

[7]While ASCE 4 does allow some variations for seismic response analysis, the principal response analysis method is basically the same as that currently used with traditional deterministic methods.
[8]Although, the initial ASCE construct was to define various SDCs for DOE facilities considering the risk consequences of a facility, this classification can also be used for individual SSCs.

- Selected regulations and guidance that are pertinent to implementation of the ASCE 43 seismic design approach were reviewed in order to identify any potential changes in current NRC regulations or guidance that may be needed.

- The LMP framework was reviewed to understand how to align the ASCE 43 seismic design approach with this initiative and with the larger RIPB framework. Two crucial factors in this alignment are to understand: (i) how PRAs will be used in making design decisions and (ii) how the sequence-level and plant-level risk measures can be integrated in an iterative design process.

- ASCE Standards 1, 4, and 43 were reviewed to understand (i) the differences in the design process when compared to current seismic design methods, (ii) the interplay among the different seismic design categories and LSs in meeting risk metrics [as opposed to the current approach of using a single seismic design category (SDC–5) and LS-D], and (iii) how the performance targets for individual SSCs could be implemented in a seismic design process.

- A stylized seven-step design process was formulated that incorporates the LMP concepts with the ASCE 43 seismic design approach in order to achieve the project objectives. This process was developed to be integrated within the broader RIPB framework and is referred to as the LMP/ASCE 43 Integration Approach. This process is described more fully in the next section of the Executive Summary.

- Several implementation issues were assessed and initial insights about how a design process could be implemented were developed. Potential impacts of the integrated seismic design process and the LMP framework on the broader regulatory and operational requirements of nuclear plant safety were also identified.

- Site-specific ground motions from nine power plant sites in the Central and Eastern United States (CEUS), representing various site conditions (i.e., hard rock, stiff soil, and soft soil sites) and geographical areas, were selected to evaluate the benefits of the ASCE 43 seismic design approach. Ground motions corresponding to the top three SDCs in ASCE 43 were compared to the design ground motions derived from current seismic design requirements to demonstrate the benefits of the enhanced RIPB framework.

- A series of simple calculations were performed to support the development of the process to demonstrate its feasibility and validity, and to better understand implementation issues and gauge the level of effort needed to conduct potential future activities to support the overall LMP/ASCE 43 integration goal. These calculations included (i) generic fragilities based on underlying assumptions used in ASCE 43 for alternate SDC and LS selections, (ii) designs of simple shear wall elements using alternate SDC and LS combinations for the same site to understand the impacts on physical designs and fragilities for these combinations, and (iii) development of a more detailed PRA-based demonstration analysis with progressive scope for future activities that includes exploring approximated adjustments to SSC fragilities to emulate different SDC/LS combinations when a detailed SPRA and supporting information are available.

Integrated Seismic Design Process (LMP/ASCE 43 Integration Approach)

A seven-step seismic design process was developed to be integrated within the broader RIPB framework. The design process builds on existing RIPB approaches in structural/seismic engineering, maintains the familiar "deterministic" process for immediate use, and utilizes existing codes and standards to the maximum extent feasible. In this seven-step process, SPRAs and seismic design are interrelated (SPRAs are currently performed in the 10 CFR Part 52 process to demonstrate plant-level seismic margins after the plants are designed). In this seismic design process, SPRAs (with appropriate risk metrics) are used to inform licensing decisions and aid the designer in assigning alternate SSC design-performance targets and design LSs. The concept of this iterative design process is to meet risk targets 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 tools during the design process itself aims to arrive at a plant-level design that is both safe and more risk-balanced, such that the SSCs margins are consistent with their risk significance within the overall system risk and performance goals of the plant and within component-level performance targets.

These seven steps of the LMP/ASCE 43 Integration Approach are:

Step 1: Select the initial ASCE 43 SDC and LS categories for each SSC and use the LBEs identified in the internal-events analysis, including the internal-events-based safety classification of various SSCs

Step 2: Design SSCs according to applicable codes for the chosen SDC/LS

Step 3: Determine the fragility of SCCs

Step 4: Perform the SPRA in accordance with applicable codes and guidance

Step 5: Check SPRA results against the F-C Target and cumulative risk criteria, as well as defense-in-depth, reliability, and other risk-informed decision-making factors. Revise SDC and LS for SSCs as appropriate.

Step 6: Repeat Steps 2 to 5, as needed

Step 7: Finalize the selection of ASCE 43 SDC and LS categories for the licensing basis seismic design

Outcomes of Current Activities Conducted to Support Project Objectives

The report provides a comprehensive explanation of the safety-focused seven-step seismic design process. The report also describes and evaluates an initial set of practical considerations that are needed to implement the LMP concepts and related RIPB improvements to the existing seismic design framework. These considerations were evaluated in the context of the proposed stylized seven-step seismic design process.

To demonstrate the potential benefits of relaxing the requirement that all safety-significant SSCs need not be designed to SDC–5 ground motions, the peak ground acceleration (PGA) and 5 hertz (Hz) ground motions for SDC–4 and SDC–3 were compared to the SDC–5 ground motions for nine sites in the CEUS. These ground motions were derived from the licensee probabilistic seismic hazard analysis (PSHA) results submittals to the NRC in response to the Commission's request based on a recommendation developed by the NRC's Near-Term Task Force (NTTF) in the aftermath of the accident at Fukushima. (This recommendation required all licensees and certain other permit holders to compute a new ground motion using the present practices and guidance and the most recent earthquake data for each site.) The procedure used by the licensee to develop the PSHA and associated response spectra were based on RG 1.208 (NRC, 2007a) and ASCE 43 (ASCE/SEI, 2005). ASCE 43 was also used in this report to obtain the SDC–3 and SDC–4 PGA and 5 Hz ground motions for these comparisons. The report provides the details of the comparative analysis. In summary, for these nine sites, SDC–4/SDC–5 average ratios for PGA and 5 Hz spectral accelerations are close to 0.55, and SDC–3/SDC–5 average ratios are close to 0.35. These results show that substantial reductions in design ground motions can be achieved through selection of alternate SDCs. The report includes comparisons of entire spectra and derives more detailed site-specific insights.

The report also includes several results that highlight the implications of some of the current regulatory requirements, such as those associated with the minimum earthquake design levels and earthquake shutdown and restart criteria. The detailed explanation of the seven steps includes discussions of additional considerations and key management and technical considerations for efficient implementation of the enhanced RIPB concepts described in the report.

The initial analyses described in the report show there are no obvious impediments in existing codes and guidance that would inhibit a more comprehensive technical evaluation or implementation of the LMP/ASCE 43 Integration Approach, including accounting for recent updates to the seismic design standards and regulatory guides. The results documented in this report will form the technical basis for detailed discussions with stakeholders (industry and other governmental agencies) at a future workshop. The goal of that workshop will be to explore the implementation and associated pros and cons of the proposed changes to the seismic design process within the larger context of seismic design, seismic safety, optimized costs, and existing seismic safety regulations.

The report also includes proposed activities to demonstrate the proposed LMP/ASCE 43 Integration Approach using available SPRA models. Based on the analyses and results in this report, the following activities should be considered for future work.

- Evaluate the seismic design of a small stylized system (going beyond a single shear wall) to more fully explore the adequacy of guidance in codes such as ASCE 43.

- Propose a cooperative effort with industry to conduct additional evaluations of the concepts and process through simplified and more detailed SPRA models. These activities will provide additional technical and regulatory insights into implementation issues and will further clarify the enhanced RIPB concepts described in this report. These cooperative activities could include:

    — An examination of a simplified and a detailed SPRA of an actual light water reactor (or advanced light water reactor) and implement the seven-step process to a practical extent to better assess advantages and limitations of the enhanced LMP approach to seismic safety.

— An examination of an advanced reactor PRA (for example, a standard modular high temperature gas cooled reactor or a fast-spectrum sodium-cooled small modular reactor) and implement the seven-step process with the goal to identify changes in current seismic safety guidance.

- Communicate the results of the current work and discuss potential future work in a workshop or a series of workshops.

Summary and Conclusions:

The following are the main conclusions from the analyses and results described in this report:

- An initial process has been developed that aligns the LMP concepts with the ASCE 43 seismic design approach, referred to as LMP/ASCE 43 Integration Approach. This is a general process that can be used for both 10 CFR Part 50 and Part 52 applications. The process is technology-inclusive.

- There are no obvious impediments in implementing the LMP/ASCE 43 Integration Approach, including accounting for recent updates to the several seismic design standards and regulatory guides. The report discusses several technical, programmatic, and regulatory considerations for the successful implementation of the framework. These discussions also form a basis for more detailed feasibility demonstration activities that may be undertaken in the future.

- The major benefits of the LMP/ASCE 43 Integration Approach come from the flexibility to assign different SDCs (different design basis ground motion levels) and different design performance limits (different LSs) to SSCs considering their risk-significance and other risk-informed decision-making considerations (in contrast to the current approach that uses a single design basis earthquake and a very stringent elastic LS for all SSCs, irrespective of their risk significance). Thus, in the enhanced RIPB seismic design process, the safety margins of individual SSCs are controlled according to their contribution to system-level and plant-level risk, thereby reducing unnecessary conservatisms (or providing additional margins, as needed) and achieving a more risk-balanced design.

- The design process of the LMP/ASCE 43 Integration Approach is very similar to that being used currently, once the SDC (design basis ground motion levels) and LS are established for SSCs. The analyses and design processes that are being used are well practiced and rely on existing standards and guidance.

- The report contributes to a technical basis to develop a regulatory guide to implement RIPB seismic design.

- Stakeholder interactions will be crucial for broader acceptance and refinement of the proposed process and for developing additional cooperative activities.

- Additional analyses are recommended to fully demonstrate the implementation and to identify associated pros and cons of the proposed changes to the seismic design process within the larger context of seismic design, seismic safety, optimized costs, and existing seismic safety regulations.

References

10 CFR Part 50. Code of Federal Regulations, Title 10, *Energy*, Part 50. “Domestic Licensing of Production and Utilization Facilities.” Washington, DC: U.S. Government Publishing Office.

10 CFR Part 52. Code of Federal Regulations, Title 10, *Energy*, Part 52. “Licenses, Certifications, and Approvals for Nuclear Power Plants.” Washington, DC: U.S. Government Publishing Office.

ASCE. “Seismic Analysis of Safety-Related Nuclear Structures and Commentary.” ASCE 4-16. New York, New York: American Society of Civil Engineers. 2017.

ASCE/SEI. “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities.” ASCE/SEI 43-05. New York, New York: American Society of Civil Engineers and Structural Engineering Institute. 2005.

Cornell, C.A. “Engineering Seismic Risk Analysis.” *Bulletin of the Seismological Society of America 58(5)*. pp. 1,583–1,606. Erratum: 59(4), 1,733. 1968.

DOE. “Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities.” DOE–STD–1020–2016. Washington, DC: U.S. Department of Energy. 2016.

INL. “Licensing Modernization Project for Advanced Reactor Technologies: FY 2018 Project Status Report.” INL/EXT-18-46151. Idaho Falls, Idaho: Idaho National Laboratory. 2018

NEI. “Risk-informed Performance-based Guidance for Non-Light Water Reactor Licensing Basis Development.” Technical Report NEI 18-04. Washington, DC: Nuclear Energy Institute, 2018.

NRC. Draft Regulatory Guide 1353 (RG-1353), “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: U.S. Nuclear Regulatory Commission. 2019.

_____. Regulatory Guide 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion.” Washington, DC: U.S. Nuclear Regulatory Commission. 2007.

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QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT

DATA: All CNWRA-generated original data contained in this report meet the quality assurance (QA) requirements described in the CNWRA QA Manual. Computations of simplified systems were implemented to examine the feasibility of integrating existing codes and standards used to ensure seismic safety with the Licensing Modernization Project initiative.

ANALYSES AND CODES: The calculations presented in this report were performed using Excel® and SAP2000 V 21.0.2. The calculations are documented in Scientific Notebook 1331E (Dasgupta, 2020).

Reference

Dasgupta, B. Scientific Notebook 1331E, “Seismic Structural Modeling and Fragility Analysis to Support Integration of Risk-Informed Performance Based Approaches to Seismic Safety of Nuclear Facilities.” San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2020.

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ABBREVIATIONS/ACRONYMS

ACI	American Concrete Institute
AEC	Atomic Energy Commission
AISC	American Institute of Steel Construction
ALWR	advanced light-water reactors
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrences
ASCE	American Society of Civil Engineers
ASD	allowable-stress design
ASME	American Society of Mechanical Engineers
BDBE[1]	beyond design basis event
CDF	core damage frequency
CDFM	conservative deterministic failure margin
CEUS	Central and Eastern United States
CFR	Title 10 of the *Code of Federal Regulations*
CNWRA®	Center for Nuclear Waste Regulatory Analyses
COL	Combined Operating License
CSDRS	Certified Seismic Design Response Spectra
DBA	Design Basis Accidents
DBE	design basis event
DC	design certification
DID	defense-in depth
DOE	U.S. Department of Energy
DRS	design response spectra
EAB	Exclusion Area Boundary
EPRI	Electric Power Research Institute
ESP	Early Site Permit
F-C	Frequency-Consequence
FIRS	foundation input response spectrum
FMEA	failure modes and effects analysis
GDC	General Design Criteria
GMRS	ground motion response spectra
HAZOP	hazard and operability studies
HCLPF	high-confidence, low probability of failure
HEP	human error probabilities
Hz	hertz
IBC	International Building Code

[1]The acronyms used in this report are based on the terms and definitions in NEI (2018). Several of these acronyms are also used for other seismic-related terminology with very different meanings. Most notably, BDBE and BDE are commonly used for *beyond design basis earthquake* and *design basis earthquake*, respectively. To avoid confusion, these terms will be spelled out in the document when the intent is related to earthquakes and not events.

IEEE	Institute of Electrical and Electronics Engineers
ISG	Interim Staff Guidance
ISRS	in-structure response spectra
LBE	licensing basis event
LERF	large early release fraction
LMP	Licensing Modernization Project
LRFD	load and resistance factor design
LS	limit states
LWR	light-water reactor
MLD	master logic diagrams
NEI	Nuclear Energy Institute
NPH	Natural Phenomena Hazard
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NSRST	Non-Safety Related with Special Treatment
NST	Non-Safety Related with No Special Treatment
NTTF	Near-Term Task Force
PGA	peak ground acceleration
PRA	probabilistic risk assessment
PSHA	probabilistic seismic hazard analysis
QA	quality assurance
QHO	Quantitative Health Objective
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RIPB	risk-informed and performance-based
RISC	Risk-Informed Safety Class
RRS	required response spectrum
SCDF	seismic core damage frequency
SDC	seismic design category
SDO	standards development organization
SF	scale factor
SLERF	seismic large early release frequency
SOV	separation of variables
SPID	Screening, Prioritization and Implementation Details
SPRA	seismic probabilistic risk assessment
SR	safety-related
SRP	Standard Review Plan
SSC	structures, systems, and components
SSDRS	Site-specific Design Response Spectra
SSE	Safe Shutdown Earthquake
TRS	test response spectrum

UHRS	uniform hazard response spectrum
U.S.	United States

1 INTRODUCTION

Over the past few years, the United States (U.S.) commercial nuclear power industry initiated the Licensing Modernization Project (LMP) to improve the regulatory basis for licensing and regulating the safety of advanced nuclear power reactors (e.g., NEI, 2018; INL, 2018). The LMP framework is supported by both the U.S. Department of Energy (DOE) and the U.S. Nuclear Energy Institute (NEI). It includes appropriate risk targets for both overall plant-wide risk and for event sequences with multiple structures, systems, and components (SSCs) and an approach to classifying safety-important 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 assuring defense-in-depth. Finally, it more directly applies probabilistic risk assessment (PRA) modeling as a basis for much of the understanding that would support safety decision-making (e.g., NEI, 2018; INL, 2018). The proposed new LMP framework relies heavily on risk-informed and performance-based (RIPB) concepts and approaches that complement the current defense-in-depth philosophy. For its part, the U.S. Nuclear Regulatory Commission (NRC) is considering adopting some or all the LMP proposals and has recently issued a draft Regulatory Guide (RG)-1353 (NRC, 2019) for public comment as part of the staff's deliberation process.

Although the LMP framework contains significant details about how its concepts should work for many aspects of reactor safety including external hazards, it does not yet explicitly address how to incorporate seismic performance criteria into the physical design of SSCs. This is particularly true for advanced reactor designs that rely on passive safety features and controls, have significantly different facility footprints, or utilize accident-tolerant fuels. The development of RIPB approaches for seismic design and seismic safety analyses in nuclear power plants date back to initiatives started in the1970s[1], and these RIPB approaches continue to evolve. This ongoing evolution of risk-informed and performance-based thinking is evident by updates to the many standards and regulatory guides that apply to seismic hazard and seismic design, such as NRC RG 1.208 (NRC, 2007a), DOE Standard 1020-2016 (DOE, 2016), and American Society of Civil Engineers (ASCE) Standards 4 (ASCE, 2017) and ASCE 43[2] (ASCE/SEI, 2005), among others.

The continuing enhancements to the RIPB concepts for seismic hazards and seismic design are also beginning to incorporate the lessons learned from recent technical activities, including work done by the nuclear industry and NRC staff to reassess seismic safety at U.S. plants based on updated seismic data, models, and methods, and considering the accident at the Fukushima Daiichi Nuclear Power Plant. For this reassessment, NRC licensees developed new ground motion response spectra (GMRS)[3] for each plant using present-day NRC requirements and guidance and compared the GMRS with the plant's Safe Shutdown Earthquake (SSE) ground motion in order to determine whether further plant risk assessments were warranted. NRC licensees then relied on seismic probabilistic risk assessments (SPRAs) to identify potential vulnerabilities given the updated seismic hazards. The SPRA results are used by licensees and the NRC staff in assessing risk-informed decisions regarding seismic safety. In new license

[1]The development of probabilistic seismic hazard analysis began in the 1960s with a seminal publication by C. Allin Cornell (Cornell, 1968).

[2]The currently published ASCE Standard 43 is Version 43-05, published in 2005. A new version is planned to be published soon. The essential philosophy of both versions is same. For the purposes of this report, ASCE 43 is cited without the year of publication.

[3]The procedure to develop the GMRS from the updated probabilistic seismic hazard curves is described in RG 1.208 (NRC, 2007a)

applications under Title 10 of the *Code of Federation Regulations* (10 CFR) Part 52, SPRAs are also required to demonstrate the overall design margin considering beyond design basis earthquakes.

Many applications of SPRA over the years have confirmed that there are significant margins in existing seismic design in-terms of ability to perform required safety functions[4]. However, some of the fundamental issues associated with the deterministic design procedures embedded in NRC's regulatory approach remain. These issues relate to not explicitly considering the behavior of safety-significant SSCs for beyond design basis ground motions and to margins that are uneven in terms of their contribution to plant-level safety or risk significance. In addition, the compartmentalized approach to design (e.g., considering each SSC separately) results in excessive conservatisms in many cases that may not contribute to overall plant safety and that do not provide a consistent way to achieve the desired seismic capability of the overall plant. In the SPRAs many of the most important event sequences involve non-seismic failures and human errors that occur in addition to failures caused by the earthquake, but current regulatory approaches do not account explicitly for those aspects of those event sequences. SPRA reflects information on the seismic capability of a plant considering aspects of design, construction, and operations. The increased use of PRAs, in general, the current movement toward a more risk-informed and performance-based licensing framework, and recent extensive experience with SPRAs provide a unique opportunity to incorporate risk informed and performance-based approaches in updates to NRC regulations and guidance that govern seismic design and seismic safety; especially in ways that lead to a more uniform, balanced, and safety- and risk-consistent plant.

1.1 Objectives

The NRC staff within the Office of Nuclear Regulatory Research (RES) initiated the project governing the work described in this report to identify and evaluate potential enhancements to the existing regulatory basis for seismic design. The objective of the project is to evaluate if and how to implement the enhanced RIPB approaches of the LMP framework together with recent improvements to industry codes and standards in nuclear seismic design (as an outcome of the evaluation, an approach referred to as LMP/ASCE 43 Integration Approach was developed). The project also seeks to identify any implementation and regulatory hurdles that could inhibit broad implementation of the LMP framework. Specifically, the NRC staff wants to develop a set of preliminary technical and regulatory insights that support (i) integration of the LMP concepts within the current NRC regulatory philosophy and regulations; (ii) adoption of appropriate risk metrics; (iii) understanding system-level performance behavior relative to the performance of individually designed components; (iv) integration of seismic-induced failures with non-seismic failures and human errors; (v) quantification of the relative risks from earthquakes compared to those from other accident initiators; and (vi) consideration of other desirable factors for overall plant safety, such as defense-in-depth and risk-balanced profiles.

These insights are intended to be technology-inclusive and explicitly account for the RIPB evaluation techniques to license future commercial advanced nuclear reactors. Any new regulations or guidance for seismic design stemming from the project should be integrated within the broader NRC RIPB framework and build on existing risk-informed approaches in structural/seismic engineering. Finally, the recommended insights should be rooted within

[4] In new license applications under 10 CFR Part 52, SPRAs are used to demonstrate the overall design margin considering beyond design basis earthquakes.

existing practices (e.g., using existing codes and standards to the maximum extent practicable) such that they can be implemented within the current regulations (e.g., 10 CFR Part 52 and Part 50) or with reasonable updates to these regulations.

The underlying strategy used to reach these objectives is to consider and evaluate the performance of individual SSCs as well as the role they play in individual event sequences; in contrast to current regulations in which every individual SSC is designed to the same seismic criteria (e.g., the SSE[5]), irrespective of the role the SSC plays in the overall system performance. This new strategy embeds risk targets into the criteria used to determine the acceptability of the seismic design of each SSC, and these risk targets explicitly account for the behavior of each important SSC when seismic input motions and loads are beyond the design basis. The underlying philosophy explicitly uses the SPRA and other considerations such as defense in depth. It allows the overall seismic design as well as the design of each individual SSC to be flexibly tailored to the desired performance in the relevant event sequences. In doing so, the strategy achieves desired safety goals and allows nuclear plant designers and operators greater flexibility in how the overall seismic design can meet system-level acceptability criteria. Thus, this strategy represents a change in the governing design philosophy: the design of each individual SSC for seismic safety would be tied to the SSC's role in the overall safety, as "measured" by the SSC's contribution to the risk of those event sequences in which it participates. Resources would be allocated where they matter the most for safety.

1.2 Scope

The scope of the project is divided in two phases:

Phase 1: Develop concepts and a process to align the LMP framework with the RIPB approach for seismic safety. This phase will also identify potential intermediate-term and longer-term activities. This Phase 1 Report will contribute to the technical basis of a future regulatory guide on the proposed approach.

Phase 2: Demonstrate the feasibility and validity of the process developed in Phase 1 through use of some simplified SPRA models and examples and obtain stakeholder input and feedback through public workshops. These activities will lead to further refinements of the concepts and process.

At later stages, a cooperative effort with the industry may be implemented in order to conduct additional evaluations of the concepts and process through more detailed SPRA models. These potential future activities can provide additional technical and regulatory insights into implementation issues and can further clarify the concepts and process. The focus of Phase 1 is to align the LMP concepts with the performance goals in existing codes and standards, especially ASCE 43, and to begin to develop recommendations for how the LMP- and ASCE-based approaches can be integrated into a consistent and practical regulatory basis for seismic design.

The work completed under Phase 1 identifies several key aspects of the broader work scope to support the broader NRC goals and objectives described in Section 1.1. An initial approach has been developed that aligns the LMP concepts with the American Society of Civil Engineers

[5]SSE is the maximum earthquake potential for which certain SSCs important to safety are designed to remain functional

(ASCE) standard ASCE 43 for seismic design. This process is referred to as LMP/ASCE 43 Integration Approach. Key aspects include:

- Applying the event-sequence-based and risk-target based philosophy embedded in the LMP framework to seismic design and defining how the LMP/ASCE 43 Integration Approach can be applied to use SSC-specific reliability or performance targets.

- Adopting the probabilistic, performance-goal-based, and graded-approach design philosophy in ASCE 43 and other supporting industry codes and standards, including ASCE 4, ASCE 7, and ANS 2.26.

- Developing and evaluating the way the LMP/ASCE 43 Integration Approach can be integrated into a consistent and practicable regulatory basis for seismic design. This aspect will be a major focus of future work.

1.3 Purpose

The purpose of this report is to describe the Phase 1 activities and to document a set of stylized fragility calculations used to describe and evaluate a set of practical RIPB-based steps that has been devised to implement the LMP concepts and related conceptual enhancements. For this purpose, a stylized seven-step seismic design process was developed, one that relies on existing codes and standards together with the application of SPRA tools during the design process itself to arrive at a plant-level design that is both safe and more risk-balanced. The concept of this design process is to meet the risk targets using combinations of variable seismic design requirements for individual SSCs and then examine their contribution to the system-level performance using the SPRA. The goal of this approach is to develop margins consistent with the risk-significance of SSCs that meet the overall system risk and performance goals of the plant, and also meet component-level performance targets. In the seven-step process, the seismic design and the SPRA are interrelated. SPRAs are currently used to support regulatory decision-making mostly to evaluate the performance and seismic risk profiles of existing, already-built, plants, and to quantify the seismic margins of those plants based on deterministic designs.

To begin to test the feasibility and applicability of the seven-step process and to evaluate its benefits, a set of fragility and seismic performance calculations were developed for two simple SSCs (a shear wall and a stylized piece of equipment). Results from these calculations are summarized in Sections 3 and 4 of this report. The exploratory analyses confirm the potential benefits that could be realized by modifying the current deterministic seismic design requirements. Most importantly, these initial calculations show that there are no obvious impediments in existing codes and guidance that would inhibit a more comprehensive technical evaluation of enhanced, risk-informed, seismic design requirements based on the LMP framework concepts and recent updates to the many seismic design standards and regulatory guides.

The purpose of the report is to also describe how several other aspects of the current NRC seismic regulatory basis can be incorporated into an approach consistent with the LMP framework. This includes minimum design requirements; requirements for restart after a large earthquake affects a nuclear plant; practical approaches for the classification of SSCs and their tailored special treatment requirements; and how seismic-margin-based logic and analysis can be used, particularly for screening or giving less emphasis to SSCs that are unimportant to risk. In addition, the results developed in this report form the technical basis for planned detailed discussions with stakeholders (industry and other governmental agencies) at a future workshop. The goal of that workshop will be to objectively evaluate the pros and cons of the proposed changes to the seismic design process within the larger context of seismic design, seismic safety, optimized costs, and existing seismic safety regulations.

2 REGULATORY BASIS

2.1 Current NRC Approach to Regulating Seismic Safety

The current U.S. Nuclear Regulatory Commission (NRC) approach to regulating commercial nuclear power plant (NPP) requires applicants and licensees to meet a suite of NRC regulations specific to ensuring that NPPs are designed and operated to perform safely during and after an earthquake. These regulations are supported by numerous regulatory guides, standard review plans, technical positions, and other regulatory documents intended to provide NRC staff with adequate guidance on how to review information submitted by these licensees and applicants and to oversee plant operations in order meet the NRC's mission to maintain public health, safety, and the environment. Many of these NRC documents refer to industry consensus codes and standards, sometimes as requirements that the NRC has adopted by reference directly into their regulations, sometimes as one (NRC-endorsed) way to meet the NRC's requirements, and sometimes by providing a methodology (a design methodology or an analysis methodology) that the NRC recognizes as adequate.

A comprehensive summary of the existing NRC reactor-safety guidance for seismic-structural evaluations was developed in NUREG/CR-7193 (Budnitz et al., 2014), to assess how applicable the existing regulatory basis is to new advanced reactor designs, especially small modular reactors. The information is summarized in a series of tables that list: (i) the relevant regulations, Regulatory Guides (RG), Interim Staff Guidance (ISG), Standard Review Plan sections, and Consensus Codes and Standards; (ii) the rationale for how these table entries were categorized; and (iii) important observations and comments on how the table entries fit within the RIPB framework or their relationship to technology-inclusive requirements. The three summary tables in NUREG/CR-7193 classify documents by whether "no changes are needed", "minor changes are needed", or "major changes may be needed" to incorporate an updated risk-informed and performance-based (RIPB) seismic safety framework. Since NUREG/CR-7193 (Budnitz et al., 2014) was published in 2014, there have been several new developments, including new Federal legislation and updated NRC and U.S. Department of Energy (DOE) documents. These are summarized in Table 2-1.

The earliest nuclear power plants in the early 1960s were designed using building codes without special seismic requirements. As the sites in California were proposed, the need for more specific considerations for seismic designs was recognized. After a significant effort starting in mid to late sixties, the Atomic Energy Commission (AEC) published the first seismic regulation as Appendix A to Part 100 in 1973. Prior to the publication of Appendix A to Part 100, the AEC published General Design Criteria (GDC) in 1971. GDC 2 requires development of a design basis for safety against natural phenomena hazards, thus requiring an explicit seismic design. The regulatory basis described in Appendix A is deterministic and prescriptive.

The seismic design uses a series of consensus codes and standards that are themselves largely prescriptive and deterministic in nature (e.g., Stevenson et al., 1984). However, from the beginning, NRC, and its predecessor the AEC, strove to keep a risk perspective in the forefront, even if at first the agency lacked any quantitative reliance on risk targets or risk analysis methods (e.g., NRC, 1986).

Table 2-1 Risk informed and performance-based developments since publication of NUREG/CR-7193

Category	Short descriptor	Name	Observations/Comments
Legislation		Nuclear Energy Innovation and Modernization Act	Directs the NRC and DOE to develop a technology-inclusive and risk-informed and performance-based regulatory basis for commercial advanced nuclear reactors.
NUREG–2213 (NRC, 2018b) NRC Office of Regulatory Research	Updated SSHAC Guidance	Updated Implementation Guidelines for SSHAC Hazard Studies	Provides updated guidance to conducting seismic probabilistic hazard analyses under the SSHAC framework based on lessons learned from recent application in the U.S. and internationally.
DOE Guidance	DOE Standard 1020	DOE-STD-1020-2016 Natural Phenomena Hazard Analysis and Design Criteria for Department of Energy Facilities	Updated guidance to provide requirements and guidance in the use of industry building codes and voluntary consensus standards in meeting Natural Phenomena Hazard requirements, particularly the International Building Code (IBC) for certain Natural Phenomena Hazard (NPH)-related situations

Through the experience of using Appendix A to Part 100, it became clear that its deterministic and prescriptive nature created ambiguities and controversies when dealing with the highly uncertain and evolving understanding of seismology. As a result, the NRC undertook several large programs in the late 1970s and 1980s to evaluate probabilistic seismic hazard analysis (PSHA) and seismic probabilistic risk assessment (SPRA). One of the most noteworthy projects was the Seismic Safety Margin Research Program (e.g., NUREG–1407 (NRC, 1991); Cummings, 1986) which developed the SPRA methodology with the experts who were in the midst of applying this methodology to address other plant licensing issues that were also uncertain and complex in nature. In recent decades, the NRC has been moving towards implementing a RIPB regulatory basis. The early efforts date back to 1994 with the Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA) described in NRC, SECY 94-219 (NRC, 1994), followed by the NRC Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (60 FR 42662, August 16, 1995), the Risk-Informed Regulation Implementation Plan [SECY 00-0062 and 00-0213, (NRC, 2000)], and, in SECY 07-0074 (NRC, 2007b).

With the advent and widening use of both PSHA and SPRA methods in the 1990s, NRC began to include performance-based elements in evaluating plant safety. Insights derived from both PSHA and SPRA studies formed the technical basis to (i) understand the actual safety levels achieved by the plants against earthquakes, (ii) determine the main contributors to the seismic event sequences that led to potential consequences, and (iii) quantify the principal uncertainties in these insights. Advances in both probabilistic hazard analyses and probabilistic fragility analyses were important contributors to improved understanding of seismic risk. In terms of regulatory requirements, these then-new PSHA and SPRA methods gradually took hold, first into some of the consensus codes and standards, and later in some of the regulations and regulatory guidance documents themselves. The DOE led the way with the publication of DOE Standard 1020-1994 (DOE, 1994), a major step forward in the development of RIPB approaches in seismic design. In this Standard, DOE evaluated the widely varying seismic risk profiles of its nuclear facilities by evaluating seismic safety within a set of seismic design categories. DOE designated four “*Performance Categories”* (PC-1 through PC-4), with the highest risk facilities, including nuclear power plants, being designated as PC-4 facilities. DOE later reclassified their facilities within five “*Seismic Design Categories*” (SDC–1 to SDC–5).

These categories were assigned to a facility, or to a safety significant structure, system, and component (SSC) within a DOE facility based on the potential severity of adverse radiological and toxicological effects on workers, the public, and the environment that could result from seismic-initiated failure. These DOE efforts led to the development of several important American Society of Civil Engineers (ASCE) standards, namely ASCE Standard 4-98 (ASCE/SEI, 1998) and ASCE 43-05, both of which have elements important to the implementation of RIPB approaches for the seismic design of SSCs within each of the DOE seismic design categories.

A major step forward in the application of RIPB regulatory philosophy took place in 1997, when NRC revised Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100 to require an evaluation of uncertainties in developing seismic design basis. The principal geologic and seismic considerations for nuclear power plant site suitability are given in 10 CFR Part 100.23, Geologic and Seismic Siting Criteria. Reviews for Combined Operating License (COL) and Early Site Permit (ESP) applications have been conducted under 10 CFR Part 52, Subpart A and associated evaluation criteria from 10 CFR Part 100.23. Specifically, Paragraph (d)(1), of 10 CFR 100.23 states, "*Determination of the Safe Shutdown Earthquake (SSE) Ground Motion requires that uncertainty inherent in estimates of the SSE be addressed through an appropriate analysis, such as a probabilistic seismic hazard analysis (PSHA)*." Guidance for the implementation of Part 100.23 and Appendix S to CFR Part 50 is provided in RG 1.208 (NRC, 2007a). A timeline depicting the development of these key standards and regulatory guides is shown in Figure 2-1.

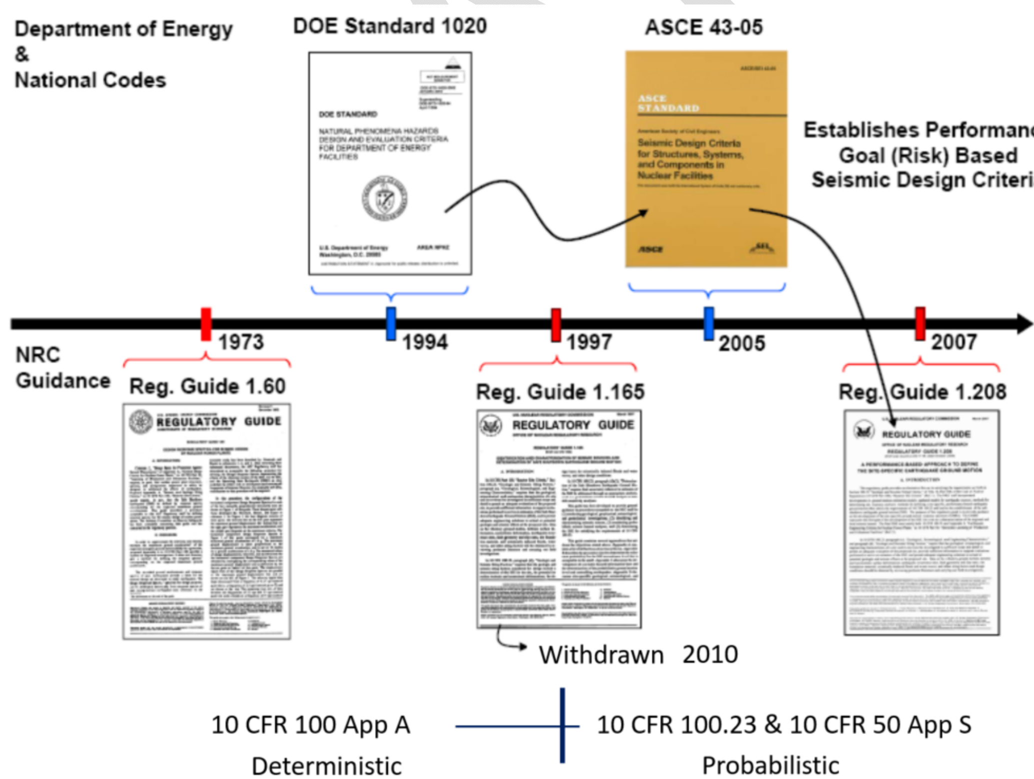

Figure 2-1 Timeline showing key regulatory guides and standards relevant to seismic safety

To a similar extent, other NRC-regulated activities regarding seismic hazards and seismic risks also moved from a deterministic approach to embrace aspects of a RIPB philosophy. These include the development of the seismic regulations and associated review plans and guidance documents for the proposed high-level radioactive waste repository at Yucca Mountain Nevada, and the use of a risk-graded approach to the development of seismic design inputs for the surface and subsurface facilities, per the requirements in 10 CFR Part 63. To meet these requirements, the DOE (as an NRC license applicant) developed performance-based criteria based on failure probabilities derived from component and system-level fragility analyses. The methodology for these fragility analyses and underlying technical bases are summarized, for example, in Dasgupta et al. (2017).

2.2 Summary of Current Codes and Standards for Seismic Design

To implement the requirements of the existing seismic regulations and requirements, the NRC, DOE and industry groups have developed a wide range of applicable guidance documents and design and construction codes, including ACI 349-13 (ACI, 2013) for reinforced concrete structures and (ANSI/AISC N690-18, 2018) for structural steel construction. Of the many applicable industry codes and standards that apply to seismic design, ANS 2.26 (ANSI/ANS, 2004), ASCE 4-16 (ASCE/SEI, 2017), and ASCE 43 are considered the most important to the LMP/ASCE 43 Integration Approach because they are overarching and incorporate many of the analysis and design requirements spread over regulatory guides and SRP sections. They address identification of target “safety levels” for a nuclear facility, define “seismic demands” for the physical design of SSCs, and identify consistent seismic design criteria for specific SSCs.

Two important considerations for seismic design of SSCs in the proposed LMP/ASCE 43 Integration Approach are the selection of the target performance category and the allowable damage state that the SSC can experience given the demands imposed by the earthquake ground motions that correspond to the target performance category. ANS 2.26 provides criteria for selecting the Seismic Design Category (SDC) for facilities and criteria and guidelines for limit states (LS) for SSCs for DOE facilities. The ANS 2.26 classifies DOE facilities in one of five SDCs based on the level of unmitigated consequences resulting from its failure: levels SDC–1 through SDC–5. These SDCs are assigned to a facility based on the potential severity of adverse radiological and toxicological effects that may result from any seismic-initiated failures. SDC–1 is for a conventional building whose failure may not result in any radiological or toxicological consequences, while an SDC–5 is the most stringent level, for example applicable to a nuclear power plant or a nuclear material processing facility with a large inventory of radioactive material. Each of these SDCs has a corresponding target performance goals, defined as the mean annual frequency of exceeding a specified limit state. Target performance goals of ANS 2.26 for SSCs are defined in ASCE 43 and are shown in Table 2-2.

Table 2-2 Target performance goals

	Seismic Design Category (SDC)			
	1 & 2	3	4	5
Target Performance Goal (P_F)/reactor-year	4×10^{-4}	1×10^{-4}	4×10^{-5}	1×10^{-5}

ANS 2.26 also provides a qualitative description of four LSs (A, B, C, or D), which characterize the limiting acceptable deformation, displacement, or stress that an SSC may experience during or following an earthquake and still perform its safety function (Table 2-3). SSCs designed to LS-A may sustain large permanent deformation (i.e., the integrity of the SSC is not essential).

Table 2-3 ANS 2.26 damage levels for each limit state

Limit State	Expected Deformation	Damage Level
A	Large permanent distortion, short of collapse	Significant damage
B	Moderate permanent distortion	Generally repairable
C	Limited permanent Distortion	Minimal damage
D	Essentially elastic behavior	Negligible damage

Acceptable damage levels for LS-B and LS-C are moderate permanent deformation and minor permanent deformation respectively. The most stringent design limits are required for those SSCs designed to LS-D, representing deformations that remain essentially elastic (i.e., the SSC is expected to return to the undeformed state after a seismic event). Each combination of SDC (SDC–1 through SDC–5) and the four LSs (A, B, C, or D) determines the design basis earthquake and acceptance criteria for design of the SSC.

The LS exceedance frequency (the expected frequency that the LS will be exceeded) is calculated by convolving a design performance[1] "fragility" curve of the SSC with the control point seismic hazard curve. The seismic hazard curves plot the annual exceedance frequency (y-axis) as a function of ground motion (x-axis) at the control point elevation. Because there is a range of spectral acceleration, seismic hazard curves are plotted for several spectral acceleration values, typically 0.5, 1.0, 2.0, 5.0, 10.0, and 20 Hz and peak ground acceleration (PGA). The design performance fragility of an SSC is defined as the probability of unacceptable performance of the SSC (i.e.., probability of exceedance of a given LS) over a range of ground motions (defined either as PGA or another specified spectral acceleration).

ASCE 4-16 provides guidance on how to evaluate seismic demands on individual SSCs in order to demonstrate that sufficient conservatism exists so that, when used in conjunction with ASCE 43, the design of individual SSCs achieves its specified target performance goal. The provisions of the ASCE 4-16 address seismic input, material properties, modeling and analytical approaches for calculating seismic demands for building structures and for developing in-structure input motions needed to design the systems and components housed within the structure. The explicit goal of the ASCE 4-16 is to provide, for each structure to be designed and analyzed, the seismic response (or seismic demand) with an 80% probability of non-exceedance for a specific seismic input. In other words, there is reasonable level of conservatism built into the procedures of ASCE 4-16 so that there is no more than a 20% probability that the computed seismic response of an SSC will be exceeded, given a specified input earthquake ground motion.

ASCE 43 describes the seismic design criteria for SSCs that are needed to ensure that the facilities can withstand the effects of the design earthquake ground motion. Specific SSCs are designed to meet target performance goals in accordance with the selected SDC and the defined LS. ASCE 43 develops a graded approach commensurate with tolerable risk. The target performance goal is expressed as an annual frequency of unacceptable performance (e.g., 1×10^{-4}/yr, 4×10^{-5}/yr, and 1×10^{-5}/yr). Unacceptable performance (i.e., a failed state)

[1] The term "fragility curve" has been used to describe two different concepts. The traditional fragility curves that are used in a SPRA relate to failure of an SSC that prohibits the performance of a required safety function. In the second concept, the LS exceedance or design performance "fragility" curves are those that represent the conditional probabilities of exceeding the design limit state for a given level of ground motion, without any consideration of functionality. See further discussion in Section 4.4.

occurs when the level of structural damage exceeds that defined by the LS (e.g., inelastic behavior of the specific SSC for LS-D).

The seismic design requirements provided in ASCE 43, in conjunction with other design, detailing, and construction standards, are considered by seismic engineers to be sufficient to meet numerical target performance goals. To achieve the target performance goals, ASCE 43 relies on the consensus codes and standards such as ASCE 4 (ASCE/SEI, 2017), ACI 349-13 (ACI, 2013) for reinforced concrete structures and (ANSI/AISC N690-18, 2018) for structural steel construction. These codes and standards produce: (i) seismic demand at 80% non-exceedance probability for the specified input and (ii) the design strength at 98% exceedance probability (i.e., a 2% probability that the design strength is less than the target). In addition, ASCE 43 aims to achieve two conditional probabilities or fragilities of the SSCs consistent with the target performance goal: (i) less than about a 1% probability of unacceptable performance for the design basis ground motion and (ii) less than 10% probability of unacceptable performance for ground motion equal to 150% of the design basis earthquake ground motion.

2.3 The Licensing Modernization Project Framework

The RIPB seismic design and the Licensing Basis Events (LBE)[2] selection process is summarized in a diagram in Figure 2-2, which process is based on the concepts in NEI 18-04. The main components of the iterative LMP RIPB seismic design process include (i) individual SSC design in accordance with ASCE 43, (ii) seismic PRA, and (iii) integrated decision making including consideration of adequacy of defense-in-depth adequacy. Within these components, plant operators and designers must (i) select the LBEs, (ii) demonstrate compliance with risk criteria, (iii) classify safety related SSCs in accordance with their risk significance, and (iv) categorize SSCs for seismic design. Of these, the demonstration of compliance with risk criteria is most novel because the LMP framework calls for a seismic design evaluation that incorporates new risk metrics incorporating event sequence frequency and public dose estimates, which differ from the traditional risk metrics of core damage frequency and large early release frequency for existing light water nuclear reactors. The iterative process shown in Figure 2-2 explicitly relies on the SPRA and aligns the LMP NEI 18-04 concepts with the ASCE 43 code for seismic design of SSCs. This process, referred in this report as LMP/ASCE 43 Integration Approach, underlines a strategy achieving desired safety goals while allowing greater flexibility in seismic design to meet system-level acceptability criteria as well as plant-level acceptability criteria.

The initial selection of operational events and internal hazards is supported by qualitative techniques such as failure modes and effects analysis (FMEA), hazard and operability studies (HAZOPs), and master logic diagrams (MLDs), and those events and hazards potentially form the basis for seismically initiated event sequences. Additional seismically induced failure modes (e.g., seismic interaction, seismically induced fire and flooding) are included in the evaluation.

Seismic event sequences include seismically induced initiating events, the plant response to an initiating event (which includes a sequence of successes and failures of mitigating systems) and well-defined end states (Figure 2-3). Each event sequence frequency is a function of the

[2]LBEs are events considered in a licensing process to derive regulatory requirements. LBEs may include normal plant operation, events anticipated to occur in the life of the facility, and off-normal events including infrequent Design Basis Events (DBEs) (ML061930123)

frequencies of the initiating event, and the reliabilities (fragilities) and capabilities of the SSCs relied on to prevent and mitigate the event sequence. The event sequence frequencies are expressed in units of events/plant-year, where a plant may be comprised of two or more reactor modules and sources of radioactive material. A generic event sequence is depicted in Figure 2-3 consisting of a seismically-initiated event and a combination of failure/success of a single SSC or a system, represented in an event tree model. For the purpose of the example in Figure 2-3 fragilities are combined at the event sequence level and the sequence level fragility is convolved with seismic hazard curve to calculate the event sequence frequency. Each event sequence, ES-1, ES-2 and ES-3, in Figure 2-3 has a specific frequency and dose magnitude.

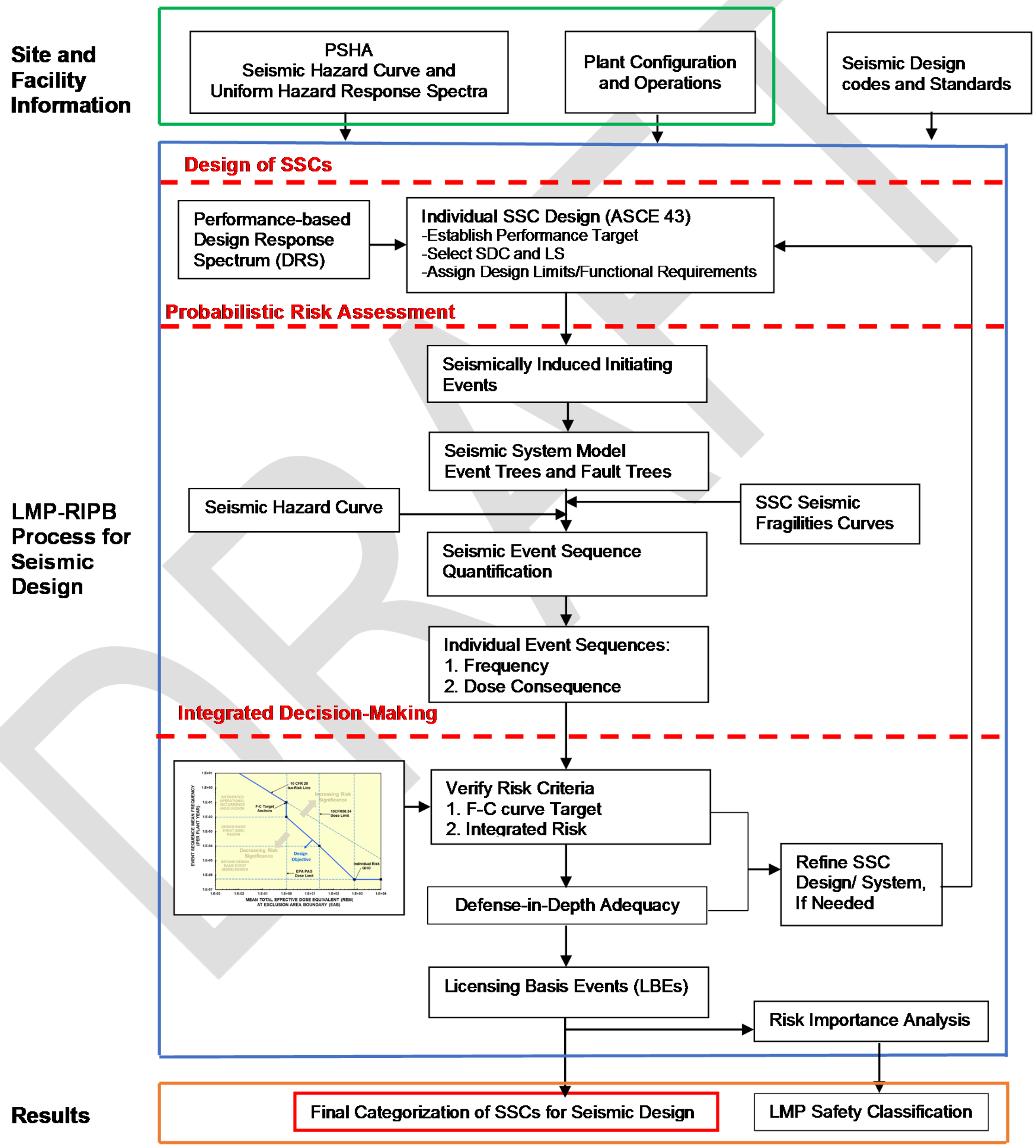

Figure 2-2 RIPB seismic design and the LBE selection, based on concepts in NEI 18-04

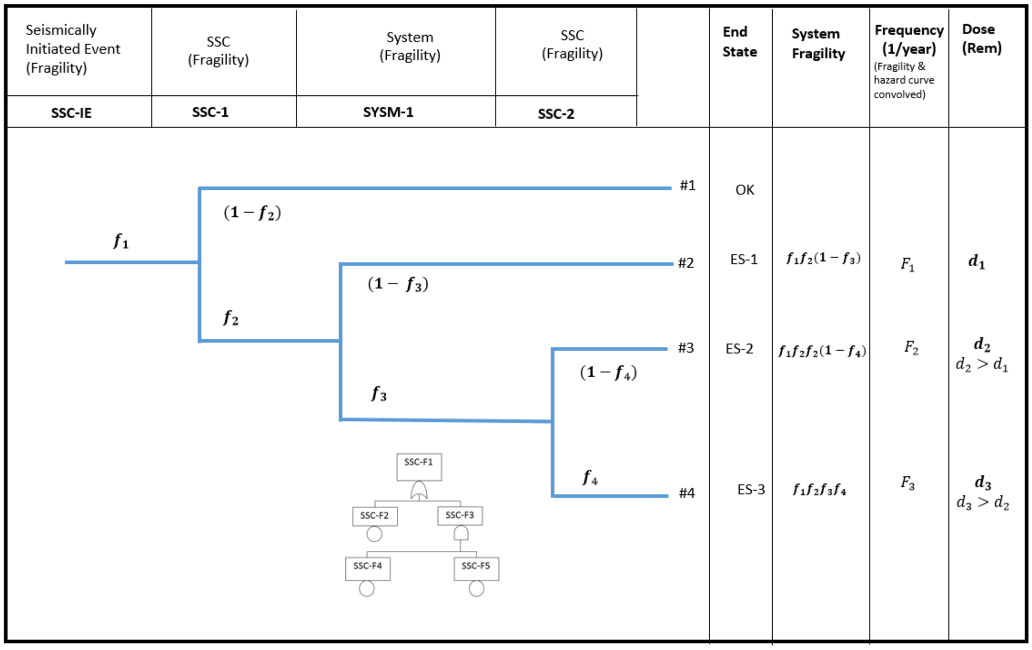

Figure 2-3 Schematic event and fault tree diagram showing seismic LBEs corresponding to an initiating event

2.3.1 Licensing Basis Events Selection Process

The LBE selection process is based on a PRA model that is intended to address the risk triplet: What can go wrong? How likely is it? What are the consequences? SPRA evaluation requires (i) a site specific seismic hazard curve; (ii) seismic fragility functions for each SSC; (iii) delineated seismic event sequences; (iv) quantification of each event sequence; and (v) estimated radiological consequences at the exclusion area boundary (Figure 2-4). The seismic hazard curve shows the annual frequency of exceedance of different spectral accelerations of ground motions (including the PGA). The fragility curve of an SSC represents the conditional probability of failure to perform the required safety function over the same range of the ground motion. The fragility function for an individual SSC is generally assumed to be a lognormal distribution function, with parameters defined by the median capacity and composite logarithmic standard deviation.

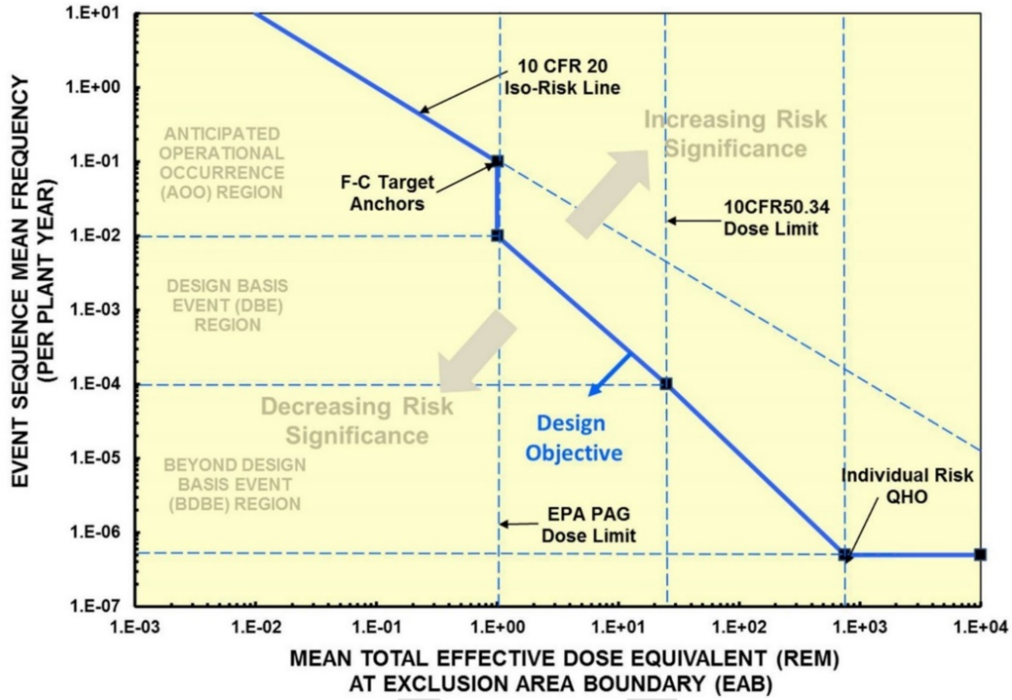

Figure 2-4 Frequency- on sequence curve (referred to as F-C Target) proposed by NEI 18-04 (NEI, 2018, Figure 3-1)

2.3.2 The Licensing Modernization Project within NRC's Risk-Informed and Performance-Based Regulatory Construct

In draft RG-1353 (NRC, 2019), the NRC staff provides guidance on using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and content of applications for non-light-water reactors, including but not limited to molten salt reactors, high-temperature gas-cooled reactors, and a variety of fast reactors at different thermal capacities. In the draft guidance, the NRC staff endorses, with clarifications, the risk-informed performance-based methodology proposed in NEI 18-04 (NEI, 2018) as one approach that can be used for the evaluation of non-light-water reactors.

The approach for licensing future nuclear power plants was first established in NUREG–1860 (NRC, 2007c) and it is intended to achieve a process that is risk-informed and performance-based, incorporates defense-in depth, and provides flexibility to the licensees or license applicants to meet safety requirements. It is important to note that under the LMP framework, the safety classification of SSCs already accounts for special treatment, such as differences in maintenance and operating requirements. However, it does not address the seismic design considerations. Thus, although the LMP framework represents a change in the governing design philosophy, the implementation of that philosophy for seismic design has not yet been written down or tested. Accomplishing such is one of the key objectives of this project, namely to propose an approach in which designing an individual SSC for seismic safety is now tied to that SSC's role in the overall safety, as "measured" by the SSC's contribution to the risk of those event sequences in which it participates. Table 2-4 describes crucial conceptual differences between the current approach for seismic design of nuclear power plants and the proposed approach based on the enhanced RIPB concepts in ASCE 43.

Table 2-4 Comparison of the current and proposed alternative approach

Current Seismic Design Approach	ASCE 43/LMP Approach
Safety Classification: All safety related systems, structures, and components (SSCs) for the seismic design purposes are considered Seismic Category 1 SSCs.	Safety Classification: Alternate safety classification is a part of the licensing modernization project (LMP) initiative considering risk significance. American Society of Civil Engineers (ASCE) 43 allows possibility of alternate seismic design categories (SDCs)[3] considering the desired level of design performance and consistent with the risk significance.
Design Basis Ground Motion: All Seismic Category 1 SSCs are designed to one ground motion level corresponding to the Safe Shutdown Earthquake (SSE) or design basis ground motion. The current site-specific design basis ground motion corresponds to the highest level determined using the Regulatory Guide 1.208 approach and it is based on the hazard exceedance frequency for a performance goal of 1×10^{-5} per year.	Design Basis Ground Motion: Design ground motions for each SDC are derived considering the performance target and margins associated with the design process. Thus, there is no single design basis ground motion for all SDC categories. The ground motions are based on hazard frequencies for targeted performance goals that can that vary with each SDC.
Design Performance Criteria: No explicit numerical criteria are defined. The design limits are associated with essential elastic behavior resulting in significant safety margins beyond the design basis ground motion.	Design Performance Criteria: A numerical design performance criterion is associated with each SDC. Alternate design limit states (LS) (e.g., allowing inelastic behavior) are permitted, considering the desired design performance and margins.
Design Procedures: The design is carried out using deterministic seismic response analyses for the SSE ground motion to establish the seismic demand. The physical design of an SSC is accomplished using established construction and engineering standards, such as those published by the American Concrete Institute (ACI), American Institute of Steel Construction (AISC), American Society of Mechanical Engineers (ASME) and others.	Design Procedures: The design employs seismic response analyses based on the ground motions that correspond to the assigned SDC categories. Seismic demands are adjusted based on the selected LS. The physical design of an SSC is accomplished using established construction and engineering standards, such as ACI, AISC, ASME and others.

Table 2-4 highlights the most important benefits of using enhanced RIPB seismic design concepts. These enhanced RIPB concepts allow flexibility to assign a different SDC to individual SSCs (i.e., flexibility in selecting the different design basis ground motion levels), and by permitting selection of alternate design LS, the desired margins can be maintained consistent with the SSC's contribution to overall plant risk. These concepts avoid excess conservatisms that do not provide commensurate safety benefits (or identifies situations where additional margins maybe needed). These concepts are practicable because once the SDC and LS are chosen, the design processes are essentially the same as those used currently[4].

[3]Although the initial ASCE construct was to define various SDCs for DOE facilities considering the risk consequences of a facility, this classification can also be used for individual SSCs.

[4]While ASCE 4 does allow some variations for seismic response analysis, the principal response analysis method is basically the same as that currently used with traditional deterministic methods.

2.3.3 Frequency-Consequence Target

NEI 18-04 (NEI, 2018) proposed a Frequency-Consequence (F-C) Target to identify acceptable accident event sequences (Figure 2-4). The F-C target is a frequency versus dose curve delineating ranges of acceptable risk for LBEs. The NEI 18-04 risk metric includes both the frequency of occurrence of the LBE sequence and the associated radiological dose to the public at the site boundary. The NEI 18-04 dose limits are consistent with NRC's Quantitative Health Objectives. In DG-1353 (NRC, 2019), the NRC staff endorse the F-C Target as a reasonable approach to determine risk significance and to support the classification of SSCs and to incorporate defense in depth. However, the NRC staff recognize that the F-C Target alone does not define strict acceptance criteria or regulatory limits. Consistent with the NRC risk-informed philosophy, including the philosophy described in RG 1.174 (NRC, 2018c), risk insights are used along with other factors for an integrated decision-making process. The F-C Target provides a general reference to assess events, SSCs, and programmatic controls in terms of sensitivities and available margins.

In addition to recommending that acceptable individual event sequences lie below the F-C Target, NEI 18-04 (NEI, 2018) accounts for aggregate risk by adding the product of the frequency and the dose of the event sequence over all LBE sequences and comparing this total risk with cumulative risk targets. If the cumulative or integrated target is not met, further enhancements to the facility design may be necessary to comply with the total risk criterion (Figure 2-2). A small set of LBEs may be identified, based on risk-information, and used to identify one or more SSCs for design enhancements. After required adjustments, the complete calculation process would be iterated until the total risk criterion is met.

2.3.4 Determination of Risk Significance and Classification of SSC in the LMP framework

The F-C Target provides a direct way to evaluate the relative contributions of risk-significant SSCs. The frequency and consequence of each individual LBE is compared to the F-C Target, as shown in Figure 2-5. LBEs with a frequency and consequence combination within 1% of the F-C Target and with site boundary doses exceeding 2.5 mrem, are considered to be risk significant. NEI 18-04 (NEI, 2018) recommends considering 95 percentile estimates of both frequency and dose to rank LBEs. In the illustration in Figure 2-5, event sequences represented by the orange dots within the shaded region of the F-C Target are considered to be risk significant. The designer has the option to further refine the design of selected SSCs for improving prevention and mitigation capabilities and increasing the LBE margin.

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 appear. Risk-insights gained from the PRA model in identifying and selecting LBEs can be used to classify SSCs. SSCs are classified as Safety-Related (SR), Non-Safety-Related with Special Treatment (NSRST) and Non-Safety-Related with No Special Treatment (NST). Safety significant SSCs include all those SSCs classified as SR or NSRST. Commonly used risk importance measures in PRA models also support risk ranking of basic events.

NEI 18-04 describes a framework that includes an integrated decision-making process where the design and risk-informed decisions are used to ensure adequacy of design and defense-in depth (DID). The evaluation includes examination of plant LBEs and SSCs relied on for prevention and mitigation of events to identify SSC capabilities and programmatic controls to support DID. The draft RG-1353 (NRC, 2019), considers the NEI 18-04 approach for assessing adequacy of DID acceptable.

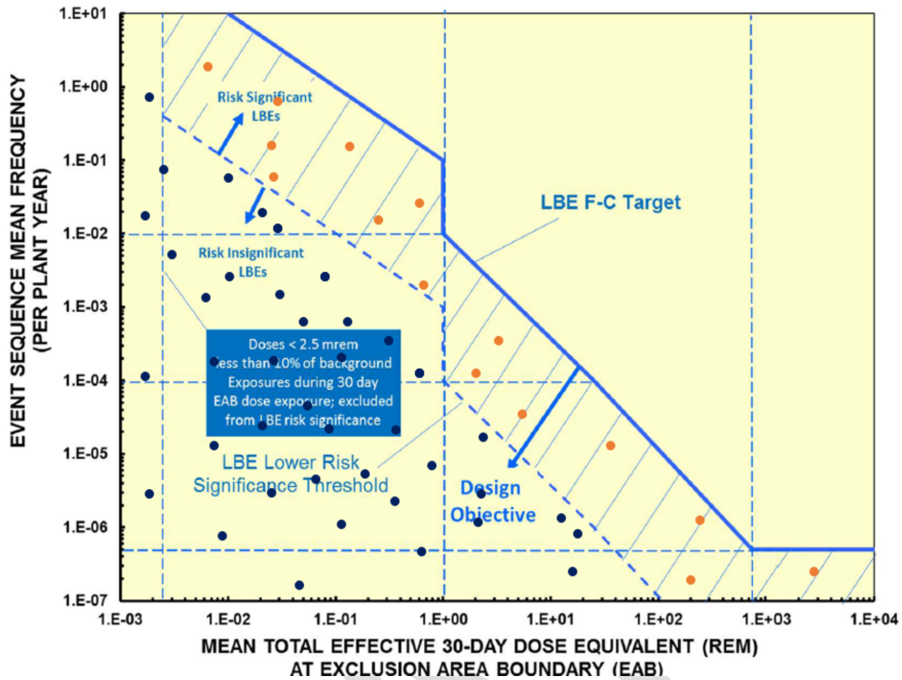

Figure 2-5 LBE Frequency-consequence proposed by NEI 18-04 showing distribution of LBEs (black dots) and risk significant LBE (orange dots)

2.4 Opportunities to Enhance the Current Seismic Regulatory Construct

As described in the preceding sections, the LMP framework establishes the conceptual foundation needed to apply the existing RIPB philosophy for the seismic design of a NPP. Moreover, many of the practical tools needed to realize these concepts in practice are available within existing codes and standards. Finally, the development and application of SPRAs has

matured to the point where they can be more fully incorporated into the design process itself. Using these concepts and tools, several opportunities exist that can enhance the current seismic regulatory construct. This subsection identifies six such opportunities that use the RIPB concepts in the LMP framework together with existing RIPB tools in existing codes and standards, especially ASCE 43, underlie the analyses provided in the next two chapters of this report. Achieving all six of these enhancements, however, is a long-term effort, and approaches for addressing all of them is not the scope of in this report. but the report is a starting point towards that goal. Issues related to the code committees (under enhancements #5 and #6) are outside the scope and purpose of this project.

1. **Enhance the current seismic regulatory construct to account adequately for the role of individual SSCs in the event sequences of most concern.**

The revolution that occurred with the advent of PRA led the safety community to understand that achieving safety requires concentrating on individual event sequences (one-by-one) and the roles of individual SSCs (one-by-one) and human errors (one-by-one) in those individual event sequences. However, the NRC's seismic regulatory philosophy concentrates almost all its requirements on assuring that each individual SSC is designed "correctly." The role of an individual SSC in various seismic event sequences is given some consideration to address some operational issues, but not at the design stage and not in a uniformly consistent way.

2. **Enhance the current seismic regulatory construct to account explicitly for the role of non-seismic failures and human errors in seismic-initiated event sequences.**

In the seismic PRAs, a significant fraction of the important event sequences involves a combination of seismic failures (beyond-design-basis failures) and one or more non-seismic failures, such as random failures of equipment to start or to run, and one or more human errors. However, the way the NRC's regulatory scheme deals with such human errors (in the control room, out in the plant, etc.) does not account for whether the human error occurs in a normal environment or in an environment during or shortly after a strong-motion earthquake has just occurred. In NRC's approach to post-earthquake human performance at the existing plants, there is, in effect, no account taken for the possible extra stress, or a more limited time period for action, or impeded access, or other issues that could accompany a large earthquake. (This situation has recently improved: In the NRC design reviews and audits for new plants reviewed recently, these issues are regularly addressed.) Neither the training nor the licensing examinations give this set of issues appropriate emphasis, and the human-machine-interface aspect of the plant design itself is not generally optimized with these considerations in mind, except in a few cases.

3. **Enhance the seismic regulatory construct to provide a mechanism for relaxing regulatory requirements for an SSC whose seismic capacity significantly exceeds what is needed to maintain system-level and plant-level safety.**

Because every SSC in the plant that is categorized as safety-related is designed to the same criteria, until the recent advent of NRC's regulation 10 CFR 50.69, no mechanism has traditionally existed for relaxing regulatory requirements for a safety-related SSC whose seismic capacity significantly exceeds what is needed. However, this relaxation basically is associated with some operational requirements and the SSCs are not redesigned. A licensee under 10 CFR Part 50.69 may be exempted for specified SSCs from certain requirements such as maintenance and inspections to reduce the operational burden. However, as the plant is already built, no changes to designs are made and the original seismic margins are retained. This one-size-fits-all approach does not allow, for example, for an SSC with lesser safety importance (but still some safety importance) to be designed to allow some modest inelastic behavior for earthquake motions at the design basis. Alternatively, for some of those SSCs perhaps the seismic input load required for the design could be smaller. Other relaxations are available that in some situations might make a major difference to facility cost, ease of maintenance, or cost of regulatory review without compromising safety. Of course, some of what underlies this issue has arisen because we need deterministic design criteria that are applicable to a variety of design situations and site conditions. Also, attention should be paid to the need for some "excess" margin because these margins may be needed in the future if the understanding of the hazard changes or if new loads are discovered.

4. **Enhance the seismic regulatory construct to be cognizant of the need to upgrade the seismic capacity of an SSC that meets current regulations but, in light of its risk significance, has inadequate margin to failure beyond the design basis ground motion.**

The NRC staff's traditional regulatory review does not distinguish between an SSC with large additional seismic margins above the design basis and one with only modest additional margins. With no way to know that, nor to know how that seismic margin "plays out" in terms of affecting overall plant risk, requiring additional margins for some SSCs is not an available option. The current "one-size-fits-all" approach is an impediment to achieving an overall balanced seismic risk profile. Although use of SPRAs as a part of the design process helps to address this issue, the overall regulatory basis would benefit if this concept was explicitly incorporated in guidance and regulation.

5. Enhance the seismic regulatory construct to acknowledge or take account of the fact that design codes for various categories of SSCs embed very different margins to failure (above the design basis)

The NRC's regulatory requirements for the seismic design of SSCs have always relied in a major way on several design codes developed for different categories of SSCs by different standards development organizations (SDOs), most of which are sponsored by professional societies. The SDOs responsible for those codes include American Nuclear Society (ANS), ASME, ASCE, ACI, Institute of Electrical and Electronics Engineers (IEEE), AISC, and a few others, and under a given SDO there can be several different code committees. Each of the code committees has always embedded extra margins in the seismic design requirements, so that an SSC designed using the code will continue to perform its function for seismic loads in excess of the design basis. However, each code committee's embedded margins are different, based on the industry practice in the individual engineering field. The result is that a nuclear power plant contains SSCs whose added seismic margins above the design basis differ from one type of SSC to another. This situation has improved in part because of the advent of probability-based design in civil engineering through "load and resistance factor design", which supplemented or replaced "allowable-stress design" years ago. For example, now all the material codes such as for concrete, steel, cold-formed and aluminum follow the same probability model in coming up with the strength factors. They may appear to be different because they properly consider the strength variability and uncertainty in the strength predictions specific to each material. The load factors and load combinations have been developed using a combination of probabilistic and deterministic bases, generated through interactions of code committees (such as ACI, AISC, ASCE and ASME) and the NRC staff over the years; they are applicable to all materials (steel, concrete, etc.) used in constructing the nuclear plants.

Nevertheless, because these codes were developed historically by different groups, and because the NRC's regulatory philosophy was historically not explicitly concerned with the details of the seismic behavior of various SSCs well beyond their design basis (even though the NRC has in recent years begun to explicitly evaluate beyond-design-basis behavior), the outcome is that there is more diversity in the embedded seismic margins than there would be if more coordination had occurred.

What this means in practice is that in an overall seismic design, insufficient account is taken for whether some of these seismic margins can be relaxed (or need to be enhanced) to lead to a more "balanced" seismic design in terms of margins that can be achieved. An overall balanced seismic design is hence more difficult to achieve. It is important to note that the recommendations in the body of this report do not rely on any changes to various industry consensus codes and standards. However, as experience is gained with the proposed enhanced RIPB approach, commensurate changes to some of those codes and standards will likely be identified, which may then further enhance the benefits that are described in this report.

6. **Enhance the seismic regulatory construct to account for how the differing margins in the various codes play out in affecting the likelihoods or consequences of individual SSC failures at the accident-sequence level.**

In the current NRC approach to regulating seismic design, the implications at the accident-sequence level of the varying design margins for different SSCs, in terms of affecting the likelihoods or consequences of an event sequence involving individual SSC seismic failures (beyond-design-basis failures), are given only limited consideration. Containment design is one example. Because of its role in preventing large releases, large margins have always been incorporated, including in the seismic design. However, whether those large seismic margins have been appropriately chosen can only be known if the containment's role in various important seismic release sequences is understood, and then that understanding is used to judge whether those seismic margins are appropriate, insufficient, or overly conservative. While this is an inevitable consequence of designing individual SSCs separately rather than designing them with cognizance of their role in the plant-as-a-whole, seismic PRAs can be used to address this issue through an iteration of the design. This philosophy is becoming the norm for the design of new plants, but this needs to be recognized more fully from the outset in any proposed revision to the overall seismic-safety regulatory construct.

In summary, this set of opportunities to enhance how the NRC approaches seismic safety regulation leads to the conclusion that, although the fleet-as-a-whole is adequately safe against earthquakes.

- the current NRC seismic regulatory construct can be enhanced in order to be consistent with the overall evolving RIPB regulatory basis;
- the current seismic regulatory construct generally produces a plant with a very unbalanced seismic risk profile;
- the current seismic regulatory construct misses out on some important methods for analyzing seismic safety that could improve our understanding; and
- the current regulatory construct also misses some opportunities for improving seismic safety itself, and for balancing cost against safety.

It is important to note that, although addressing these several opportunities will take some time, there are no known impediments to implementing them except the usual (and important) difficulties of obtaining consensus among all interested parties and assuring that new approaches as they come into the overall seismic regulatory construct preserve the consistency and usefulness of the system now in place.

3 INCORPORATING THE ENHANCED RIPB CONCEPTS IN THE SEISMIC DESIGN PROCESS

3.1 Background

Risk-informed, performance-based (RIPB) approaches are being used in several applications for operating reactors and other nuclear facilities as discussed in Chapter 1. These applications include development of site-specific seismic ground motions for design using probabilistic seismic hazard analysis (PSHA) in conjunction with probabilistic criteria. They also include seismic probabilistic risk assessments (SPRAs) to address both plant-specific issues and generic issues, and they include currently accepted alternative RIPB regulatory approaches, such as those in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.69. At various stages during licensing, such as design certification, application for a combined license to build and operate a nuclear power plant (NPP), and verification of seismic margin capacity prior to fuel loading, the new plant designs require a probabilistic risk assessment (PRA) margin analysis, or an SPRA to demonstrate that the performance target is met (NRC, 2010a).

Many of these applications are geared toward understanding the overall integrated plant response to emerging issues, such as updated seismic ground motions, or issues related to operating experience. What distinguishes the proposed seven-step seismic-design process and LMP/ASCE 43 Integration Approach discussed in this report is the incorporation of enhanced RIPB concepts in the actual seismic design itself, where the seismic hazard levels used in the design are chosen for each safety-related structure, system, and component (SSC), commensurate with the SSC's contribution to risk. This seven-step process would lead to seismic margins consistent with the risk-significance of SSCs within the overall risk and performance goals of the plant and within component-level performance targets. This proposed seven-step process is in contrast to the application of a single hazard level for the design of all safety-related SSCs and for the entire facility, and in contrast to current practice, where PRA methods are primarily used to evaluate the performance and risk profiles of existing plants, and to quantify seismic margins of those plants based on their original design.

The goal of the proposed seven-step process is to evaluate the applicability of enhanced RIPB concepts within the design development itself. A proposed seven-step process integrates the existing seismic-safety RIPB philosophy with the licensing modernization project (LMP) initiative concepts (Chapter 2) to achieve greater safety and cost benefits.

3.1.1 Guiding Principles

In developing the seven-step process discussed here, the following guiding principles were considered

- Integrating seismic RIPB concepts within the LMP framework
- Building on existing RIPB approaches in structural/seismic engineering;
- Recognizing that the actual design process itself is still fundamentally "deterministic";
- Utilizing existing codes and standards to the maximum extent feasible; and
- Identifying and suggest updates to the regulatory basis and guidance, as necessary.

Chapter 2 described the general LMP framework and the considerations involved in integrating the external hazards for defining Licensing Basis Events (LBEs) and SSC safety related classifications. Initially, it is expected that the LBEs and SSC safety classifications for the design be likely based on a PRA that covers internal events only. For those external hazards amenable to PRA methods including seismic hazards, there may be new LBEs created that represent the event sequences initiated by the earthquake ground motions and other external-hazard loads. This would include design basis events and beyond design basis events induced by external hazards modeled in the PRA and would be subject to evaluation using the Frequency-Consequence (F-C) and cumulative risk targets. For some SSCs, the initial SSC classifications may likely change. Design is inherently iterative[1] and will require a few iterations to establish the seismic design categories for the design of individual SSCs. The proposed integration in this chapter is to have a process that defines performance targets for individual SSCs such that the overall performance targets, both F-C dose and cumulative targets, are met. As described in Chapter 2, considerable infrastructure and experience exist in the fields of seismic design, evaluations, and PRA. The intent is to build on this experience.

In ASCE 43[2] (ASCE/SEI, 2005), the risk-informed and performance-based considerations are achieved through developing a design basis ground motion from a PSHA with a specified annual frequency of exceedance in accordance with the chosen seismic design category (SDC), with a specific probabilistic performance target, and the selection of a design limit state (LS). The compliance of the overall design to broader performance targets is demonstrated through an SPRA. Once the design basis motion is established, the seismic response analysis and design procedures basically can be considered “deterministic” reflecting very well-established criteria and practices. Although, alternative approaches are allowed in ASCE 4 (ASCE/SEI, 2017) and ASCE 43, that would use more explicit probabilistic approaches, there is no substantial experience using these more probabilistic approaches in design. Therefore, discussion here is limited to the deterministic design practice option of the ASCE 4 and 43 codes. It is important to retain this deterministic framework and use the existing construction codes, such as American Concrete Institute codes, for the actual design of components. This is the only practical approach for designing a complex facility such as a nuclear power plant using large teams involving various engineering disciplines.

In developing the seven-step process, consideration was given to differences that may arise with respect to the existing guidance, practice, and regulatory construct. No impediment that will preclude the process has been identified. The discussion includes issues that require consideration during the implementation stage. Several such situations are identified and discussed in Section 3.3.

3.1.2 Key Assumptions and Considerations

In current NRC regulations the safety related SSCs and certain safety important equipment (e.g., spent fuel pool racks) are denominated Seismic Category 1 SSCs and they all are designed to the same design basis earthquake ground motions. These SSCs also include SSCs that are not related to the reactor accident risk. For example, risks associated with potential accidents in spent fuel pools are analyzed through a separate PRA in the LMP framework. This

[1]This is a reasonable expectation, as all design processes are iterative to some extent, but the proposed process will add additional iterations to establish SSC design categories.
[2]The currently published ASCE Standard 43 is version 43-05, published in 2005. A new version is planned to be published soon. The basic philosophy of both versions is same. For the purposes of this Chapter, ASCE 43 is cited without the year of publication.

separate PRA also includes seismic initiated event sequences. The SSCs related to non-reactor radiological sources, such as radiological waste hold-up tanks, are designed to lesser design requirements, but they are within the purview of the NRC reviews and safety evaluations. Similarly, NRC reviews also consider the potential for failures of non-safety related SSCs to adversely impact safety-related SSCs.

The following four assumptions are used in describing the proposed seven-step process.

1. For risks resulting from reactor operations, two LMP safety classifications, safety related (SR) and non-safety related with special treatment (NSRST), are assumed to be within the scope of the NRC review and their designs will comply with the NRC regulations, accepted guidance, and industry codes and standards.

2. SSCs, such as waste holding tanks and spent-fuel pools, are also assumed to be designed using the process described here, as the LMP framework considers the risk arising from all sources of radiological hazards and all plant operating modes.

3. Although many combinations of SDC and LS are possible (e.g., SDC–5 with LS-A, SDC–5 with LS-B, SDC–5 with LS-C, etc.), the number of combinations that are practical are assumed to be limited because of technical and regulatory considerations. These considerations are discussed in detail in Section 3.3. For the analyses provided in this report, only combinations of SDC–3, SDC–4 and SDC–5, with LS-C and LS-D were used.

4. Licensing according to 10 CFR Part 52 incorporates three stages: (i) certified design; (ii) combined license application to build and operate at a site; and (iii) the U.S. Nuclear Regulatory Commission (NRC) approval prior to fuel loading. A similar structure may be used for the reactor design and licensing that implement the LMP framework. The proposed seven-step process assumes this this three-stage structure and identifies several specific licensing considerations, discussed in this report. However, the proposed seven-step process is equally applicable to other regulatory structures, or to a site-specific design and licensing process that would occur under 10 CFR Part 50.

3.1.3 Nomenclature

Several design basis ground motions are possible under the seven-step process. Under 10 CFR Part 52, there is the design basis ground motions for the certified design and a site-specific design basis ground motion to account for site-specific hazards. For advanced reactors, the terminology of the Safe Shutdown Earthquake (SSE) may be confusing as the required safety functions discussed in Chapter 2 do not explicitly include reactor shutdown. ASCE 43 uses both the terms “design basis earthquake” and “design response spectra” (DRS). ASCE 43 further states that, “the design basis earthquake ground motion shall be defined in terms of the Design Response Spectra (DRS).” In addition, the current terminology is counter intuitive compared to the use of “Seismic Category 1” for safety related structures. The nomenclature provided in Table 3-1 is used in this report to be consistent with the terminology in the LMP framework and with the safety functions of advanced reactors. In order to avoid confusion, and In cases where the terminology differs from the LMP framework, those difference will be explicitly identified.

Table 3-1 Seismic design nomenclature

ASCE Seismic Design Category	Certified Seismic Design Response Spectra (CSDRS)	Site-specific Design Response Spectra (SSDRS)
SDC–5	CSDRS-5	SSDRS-5
SDC–4	CSDRS-4	SSDRS-4
SDC–3	CSDRS-3	SSDRS-3

3.2 Overview of Seven-Step Seismic Design Process (LMP/ASCE 43 Integration Approach)

Although ASCE 43 has been used to evaluate the seismic safety of existing nuclear facilities, it has never been used in the actual design of nuclear power plants. Nevertheless, the ASCE 43 design response analysis and strength design process for SDC–5 and LS D reflect much of the current design practices used for light-water reactors (LWRs) and is consistent with the current staff positions in Regulatory Guide (RG) 1.208 (NRC, 2007a). The proposed seven-step process builds on insights gained from the application of ASCE 43 to safety evaluations of existing facilities and from the design practices for current LWRs.

ASCE 43 SDCs and LSs graded approach require definition of performance goal for different SSCs, which cannot be derived solely from the Frequency-Consequence (F-C) plot of the NEI 18-04 document (NEI, 2018). There is a multitude of SSCs in the various event sequences, and hence there are multiple possible performance combinations of the individual SSCs yielding the same overall performance of the event sequence. Therefore, one potential approach is to use predetermined SSC categories and performance goals and then rely on the PRA to demonstrate how close the resulting F-C pairs are to the F-C Target and how the design meets the F-C limits for the individual event sequences and the cumulative risk metrics. This is an inherently iterative process that could also lead to identification of additional LBEs and the recategorization of SSCs. The risk target can be achieved by re-designating the safety classification, selectively hardening or relaxing the design, introducing redundancy, improving random failure rates, or improving human-error probabilities, or some combination. Approaches to demonstrate the feasibility of this approach are discussed in the next chapter. A more detailed feasibility analysis and application demonstrations are proposed for future activities.

The seven-step seismic design process is illustrated in Figure 3-1. A more detailed integrated process is shown in Figure 2-2, which also illustrates how this process fits into the overall LMP framework. Section 3.3 provides a more detailed description for each of the steps, including technical considerations involved in implementing these steps. These steps are complex, and each step involves several decisions and intricate activities. While the proposed seven-step process is for advanced reactors, the concepts are also applicable to advanced light-water reactors (ALWRs) or LWR designs with some specific modifications and considering associated risk metrics. A modified process for LWRs (or ALWRs) is described in Section 3.4.

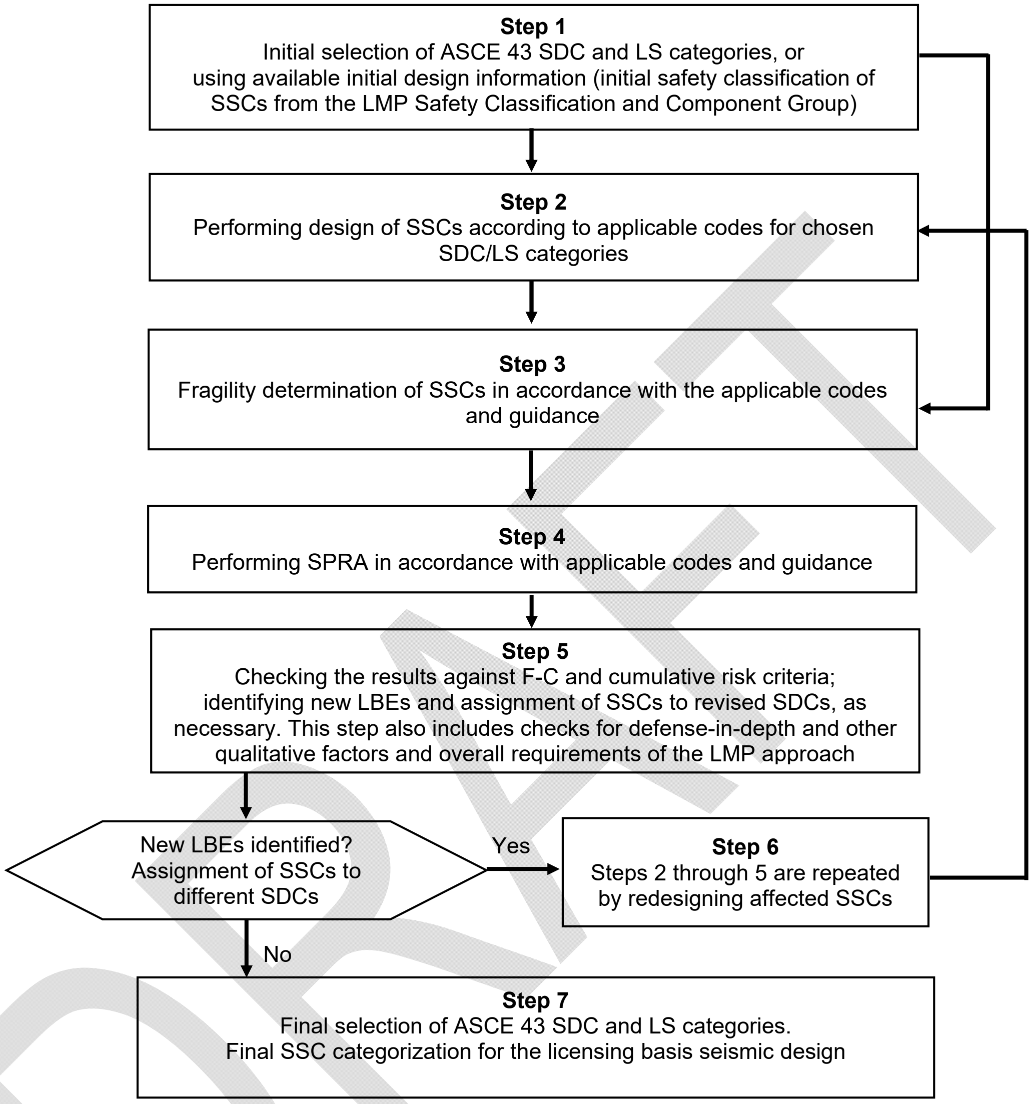

Figure 3-1 Proposed seven-step seismic design process (LMP/ASCE 43 integration approach)

The proposed seven steps of the LMP/ASCE 43 Integration Approach are described as follows:

Step 1: Initial Selection of the ASCE 43 SDC and LS Categories

The design LMP Safety Classification and Component Group (LMP Component Group)[3] establishes an initial classification of SSCs based on an internal events PRA. For LWRs, this step may be relatively straightforward because of existing designs and past design experience.

For advanced reactors, whose designs are already in progress or have been completed, the seismic design is based on an approach akin to ASCE 4 and ASCE 43 using SDC–5 and LS-D

[3] The "LMP Component Group" has overall responsibilities for establishing LBEs and SSC classification and applies the integrated decision-making process of the LMP framework. This is a multi-disciplinary group covering pertinent technical and regulatory expertise.

requirements. 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 are selected at the onset.

Additional considerations on the initial selection of SDC and LS categories include regulatory requirements, stability of the designs, and available information. Section 3.3 of this report provides additional details describing the technical and regulatory considerations used to develop the initial selection of SDCs and LSs.

<u>Step 2: Seismic Design</u>

In most cases, the initial design would adopt SDC–5 and LS-D. For the subsequent iterations in the evaluation of risk, alternative combinations of SDCs and LS may arise. For advanced reactors, it may be possible to implement seismic design according to ASCE 43, 4, 1, applicable NRC industry guidance, and other codes and standards according to selected SDC and LS categories in Step 1. However, in the absence of the detailed design, the original starting design may best be limited to SDC–5 and LS-D.

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

In consultation with the LMP Component Group and in accordance with the applicable ASME/ANS SPRA standard, it will be necessary to determine the fragilities of the identified SSCs.. In developing these fragilities, the purposes of conducting a SPRA, to determine the design criteria for various safety related SSCs, should be considered.

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

Using the fragilities developed in Step 3 and the SPRA models developed in accordance with the applicable codes, this step performs an SPRA. The LMP approach requires an SPRA at the stage of design certification. This is a departure from the current approach being used in the design certification of ALWRs. As described in SECY-93-087 (NRC, 1993) and ISG-020 (NRC, 2010a), a PRA based margin approach, which does not require a PSHA, is used to demonstrate whether the proposed design meets a plant-level performance target of high-confidence, low probability failure (HCLPF) level of 1% at 1.67 times the design basis ground motion. The plant-level HCLPF in this context is with respect to the PRA's core damage frequency (CDF) and large early release frequency (LERF). In other words, this performance target assures that seismic induced core-damage or large early releases are not likely to occur at the ground motion levels that are 1.67 times the design basis ground motion. In contrast, the LMP framework requires a SPRA for the purposes of establishing LBEs and the performance targets are in terms of frequencies and consequences. Approaches are needed to select seismic hazard information that can be used for a design certification SPRA.

After completing the SPRA, results are provided to the LMP Component Group to check whether the results meet the various risk metrics including the cumulative risk metric.

<u>Step 5: Check the Proposed Classification Against the Risk, Defense-In-Depth, and Other Qualitative Criteria</u>

The LMP Component Group evaluates the results of the initial PRA to determine whether the individual event sequence risks are within F-C Target limits, whether the integrated risk criteria are met, and which risk significant LBEs fall within 1% margin on the F-C curves. This group also evaluates defense-in-depth adequacy, reliability, and other qualitative factors related to the risk-informed decision-making (e.g., balance between prevention and mitigation, 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 in order to meet the desired safety and cost goals, and all regulatory requirements. This final design, incorporating the various combinations of SSCs with different SDC and LS assignments, may be different than the initial classification developed in Step 1.

Step 7: Final SSC Categorization for Seismic Design

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

3.3 Detailed Discussion of the Seven-Step Seismic Design Process

This section discusses additional considerations in each step to effectively implement the seven-step design process. These considerations are important to maintain a stable design process, which is a process that results in a seismic design of a nuclear power plant (at the design certification stage) that is robust enough to be placed at multiple sites and requires minimal site-specific changes. However, at the same time, the intent is to optimize the selection of seismic design categories and design limit states to achieve the maximum benefits of the RIPB approaches.

In the following discussion, it is important to note that the CSDRS are site parameters assumed for the design certification (DC) design under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52 process. The design vendor selects CSDRS and associated generic site profiles for seismic safety analyses. The purpose of this section is to show that the RIPB process can be implemented in the context of the 10 CFR Part 52 process.

3.3.1 Step 1—Initial Selection of the ASCE 43 SDC and LS Categories

Ground rules for choosing the SDC and LS categories should be established early in the process. For example, in situations where a SSC may provide structural support to multiple other SSCs in lower categories, the supporting SSC could be analyzed and designed to a hazard level (seismic input) higher than required for the supported lesser category SSCs (e.g., the supporting SSC is designed as SDC–5 and the supported SSCs may be designed as SDC–4). An estimation of seismic loads for supported SSCs based on SDC–4 category would be required in this case.

Two options are presented for selecting initial SDC and LS categories. Option 1 starts with selection of SDC–5 and LS-D for all four LMP safety classifications. Option 2 allows less stringent SDC and LS categories. As an example, Table 3-2 represents selections of SDC/LS pairs for both options. Two other “safety-related” categories are added to Table 3-2 to reflect non-reactor radiological sources and other SSCs that may not be part of any risk assessment.

For the reasons discussed later in this section, the initial selection is limited to SDC–4 and SDC–5 and LS-D and LS-C. The table also includes the nomenclature of design basis ground motion corresponding to the selected SDCs discussed in Section 3.1.3.

In the Table 3-2, SDCs and LSs are assigned to a group or a class of components. Although, it is feasible to establish an individual SDC and LS classification for each SSC, the process can become unwieldy and cumbersome. However, some major components may be assigned independent SDC and LS categories considering potential cost and safety benefits.

Table 3-2 Initial selection of SDC and LS categories

Safety Categories	Option 1 ASCE SDC and LS	Option 2 ASCE SDC and LS
Structures, systems, and components (SSCs) selected by the designer to perform required safety functions to mitigate the consequences (of DBEs) to within the Frequency-Consequence (F-C) Target, and to mitigate DBEs to meet the dose limits of 10 CFR 50.34 using conservative assumptions.	SDC–5 LS-D CSDRS-5 (or SSDRS-5)	SDC–5 LS-D CSDRS-5 (or SSDRS-5)
SSCs selected by the designer to perform required safety functions to prevent the frequency of DBEs with consequences greater than 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C Target.	SDC–5 LS-D CSDRS-5 (or SSDRS-5)	SDC–5 LS-D CSDRS-5 (or SSDRS-5)
Non-safety related SSCs relied on to perform risk significant functions. Risk significant SSCs are those that perform functions that keep licensing basis events (LBEs) from exceeding the F-C Target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs.	SDC–5 LS-D CSDRS-5 (or SSDRS-5)	SDC–5 LS-C (or C SDC–4 LS-D) CSDRS-5 or CSDRS-4 (SSDRS-5 or SSDRS-4)
Non-safety related SSCs relied on to perform functions requiring special treatment for defense-in depth (DID) adequacy.	SDC–5 LS-D CSDRS-5 (or SSDRS-5)	Review current approaches Allow an applicant to choose SDC and LS SDC no less than 4?
All other SSCs except for those included in the row below	Use currently practiced approaches Allow an applicant to choose	Use currently practiced approaches Allow an applicant to choose
Not included in seismic probabilistic risk assessment (SPRA) models, but radiological sources (e.g., spent fuel pool)	Use current approaches as follows SFP SDC–5, LS-D for spent fuel pool	Use current approaches as follows SFP SDC–5, LS-D for spent fuel pool

3.3.1.1 Considerations Related to Choices of SDC Categories

As discussed in Chapter 2, in ASCE 43, the annual frequency of exceeding the acceptable performance level is called the target performance goal, and it decreases with increasing SDC level (Table 3-1). This decrease in the target performance goal is achieved by decreasing the annual exceedance frequency of the design basis earthquake ground motion (i.e., going from SSDRS-4 to SSDRS-5). One important consideration in selecting an SDC is understanding the relative differences among the design basis earthquake ground motions. For reference, the DRSs[4].associated with the SDC–3, SDC–4, and SDC–5, are derived from the uniform hazard response spectra (UHRS) with annual frequency of exceedance of 1×10^{-3}, 4×10^{-4}, 1×10^{-4}, and 1×10^{-5}, respectively.

In order to evaluate the differences among the SDCs, several recent site-specific PSHA results PGA and the 5 hertz (Hz) spectral frequency are summarized in Table 3-3. The ground motions were derived from recent licensee submittals in response to Fukushima Near Term Task Force (NTTF)

Table 3-3 Ground motion data from nine CEUS sites

Plant Site	General Site Characteristics	1×10^{-4}		1×10^{-5}		GMRS
		PGA (g)	5 Hz (g)	PGA (g)	5 Hz (g)	PGA (g)
A	Rock	0.325	0.313	1.05	0.983	0.499
B	Rock	0.378	0.328	1.201	1.01	0.572
C	Soil over rock	0.436	0.876	0.814	1.77	0.436
D	Soil over rock	0.153	0.223	0.389	0.559	0.194
E	Soil over rock	0.207	0.49	0.521	1.13	0.26
F	Till over rock	0.34	0.351	1.06	1.32	0.505
G	Soil	0.12	0.242	0.352	0.737	0.17
H	Soil	0.069	0.165	0.182	0.394	0.090
I	Soil	0.089	0.161	0.219	0.372	0.11

Recommendation 2.1. Figure 3-2 shows mean PGA hazard curves for the sites at the SSE control point locations. These sites are Central Eastern United States (CEUS) nuclear power plant locations situated in different physiographic regions with different site characteristics.

Current NRC guidance uses the ASCE 43 SDC–5 approach to develop the ground motion response spectra (GMRS), as described in RG 1.208 (NRC, 2007a). These comparisons are useful to understand differences in the resulting ground motions for alternate SDC categories, and to compare to the current practice.

[4]The SSDRSs are derived from the newer procedure for developing the design response spectra (DRS) that will be included in the new version of ASCE 43. However, the results are very similar to the DRS derived from the procedure outlined in ASCE 43-5. This procedure is illustrated in Section 4.2.2 of this report. The annual frequency of exceedance of 4×10^{-4} is included here as ASCE 43-05 uses this UHRS to derive DRS for SDC–4. Similarly, the annual frequency of exceedance of 1×10^{-5} is included as the revised procedure to develop DRS for SDC–5 uses this UHRS.

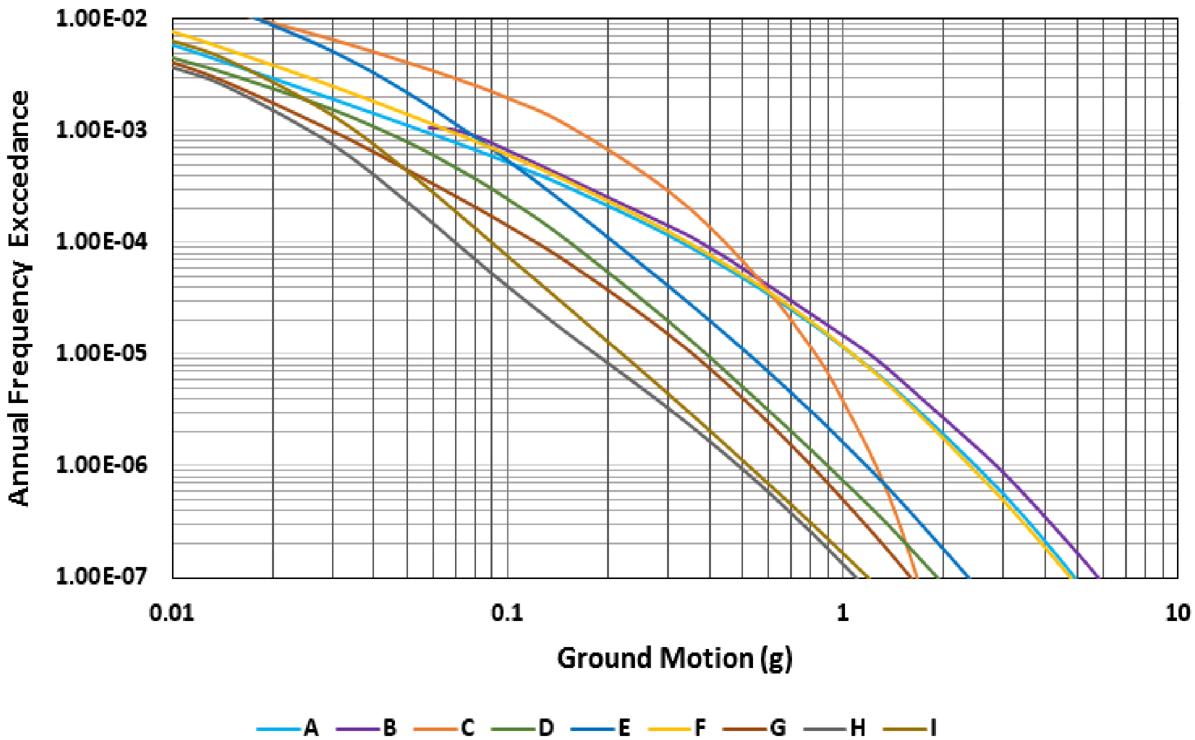

Figure 3-2 Mean hazard curves (PGA) at the control points for the nine CEUS sites described in Table 3-3

Because the establishment of the DRS in ASCE 43 is based on modifying the underlying UHRS, it is instructive to examine how the UHRS compares for different exceedance frequencies and different site conditions. Figure 3-3 through Figure 3-6 show UHRS for all sites at exceedance frequencies of 1×10^{-3}, 4×10^{-4}, 1×10^{-4}, and 1×10^{-5}. Figure 3-6 is included because the revised process proposed in the upcoming edition of ASCE 43 uses UHRS associated with 1×10^{-5} annual exceedance frequency to derive the DRS for SDC–5.

These figures also include a plot of the RG 1.60 (NRC, 2014) spectrum anchored at 0.1g PGA. The purpose of including the RG 1.60 spectrum is to understand how the design basis ground motions associated with various SDC categories compare with the current regulatory requirements for a minimum earthquake level. Appendix S to 10 CFR Part 50 states *"The horizontal component of the Safe Shutdown Earthquake Ground Motion in the free-field at the foundation level of the structures must be an appropriate response spectrum with a peak ground acceleration of at least 0.1g."* NRC ISG-017 (NRC, 2010b) further states that the RG 1.60 spectra are considered appropriate spectra for meeting this part of the regulation. ASCE 43 includes similar statements: *"If required, the DRS shall be amplitude scaled up by one factor across the entire frequency range such that the zero period acceleration is not less than 0.04 g for SDC–2, 0.06 g for SDC–3, 0.08 g for SDC–4, or 0.10 g for SDC–5."*

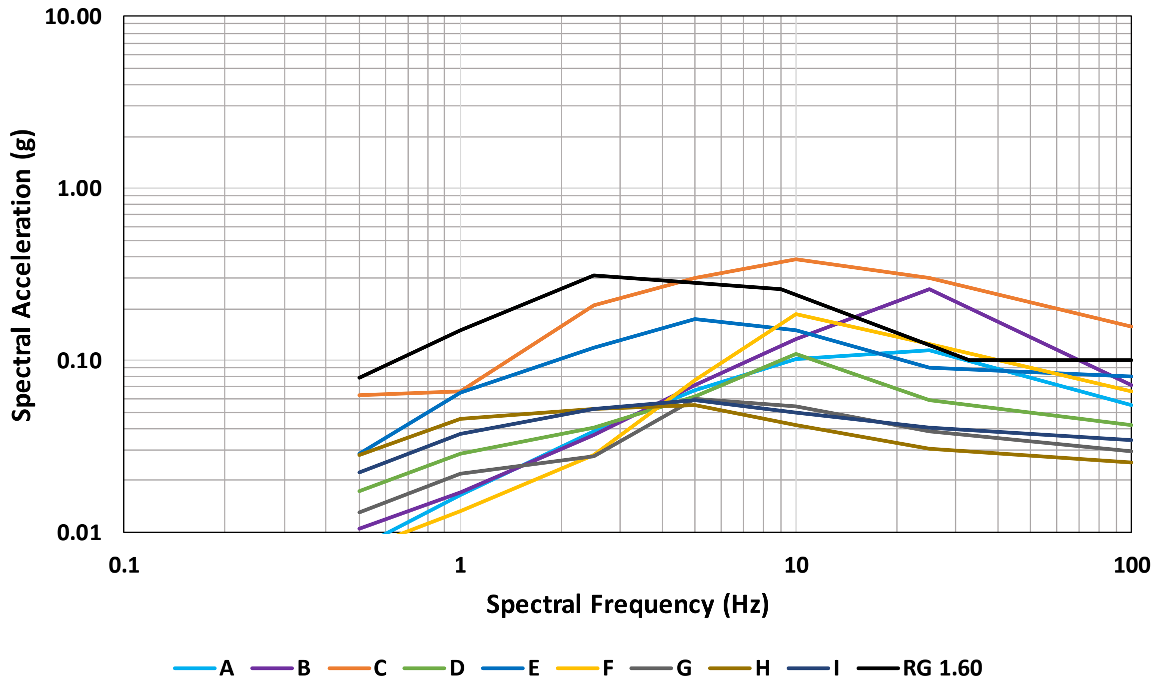

Figure 3-3 UHRS for nine CEUS sites corresponding the $1{\times}10^{-3}$ exceedance frequency.

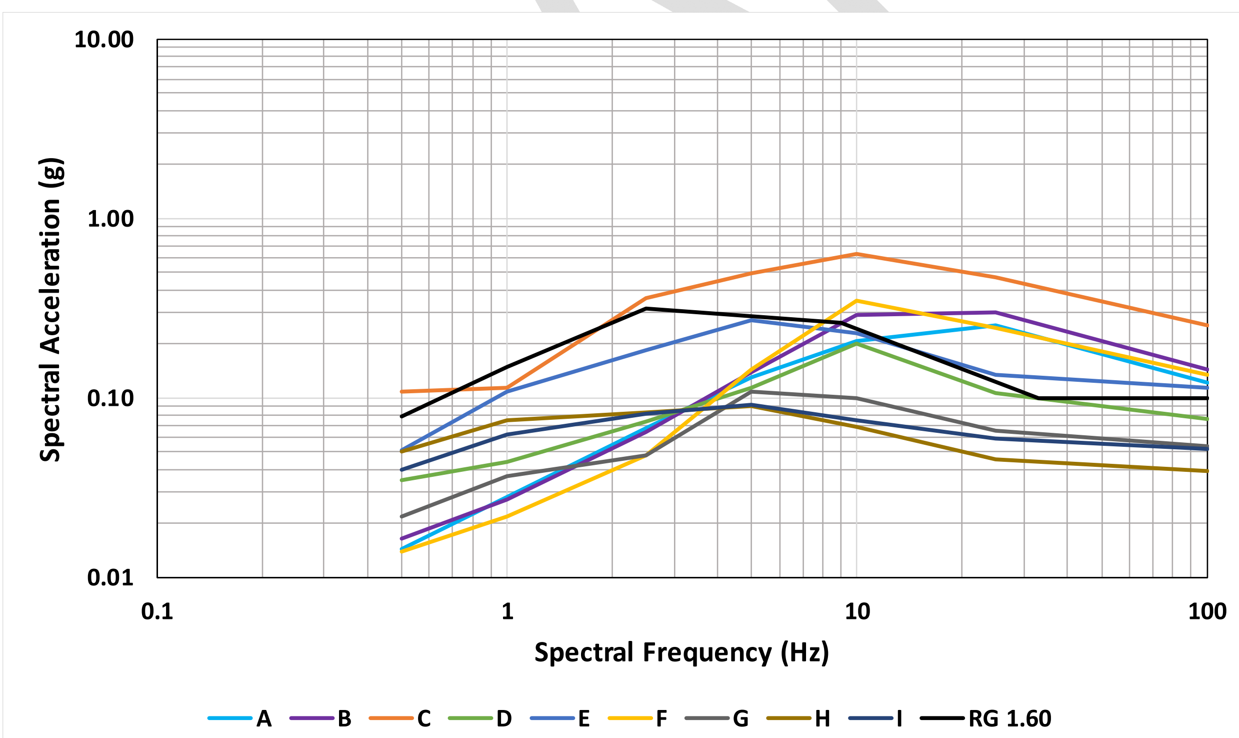

Figure 3-4 UHRS for nine CEUS sites corresponding the $4{\times}10^{-4}$ exceedance frequency.

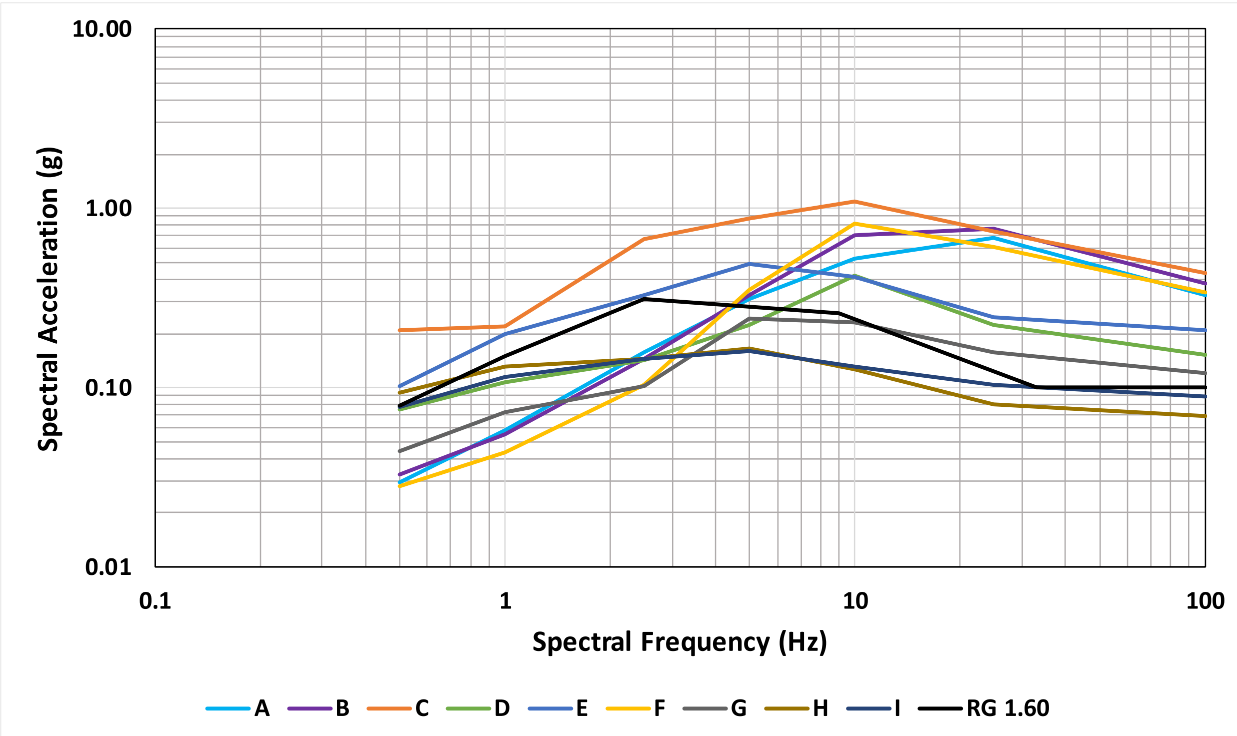

Figure 3-5 UHRS for nine CEUS sites corresponding the 1×10^{-4} exceedance frequency

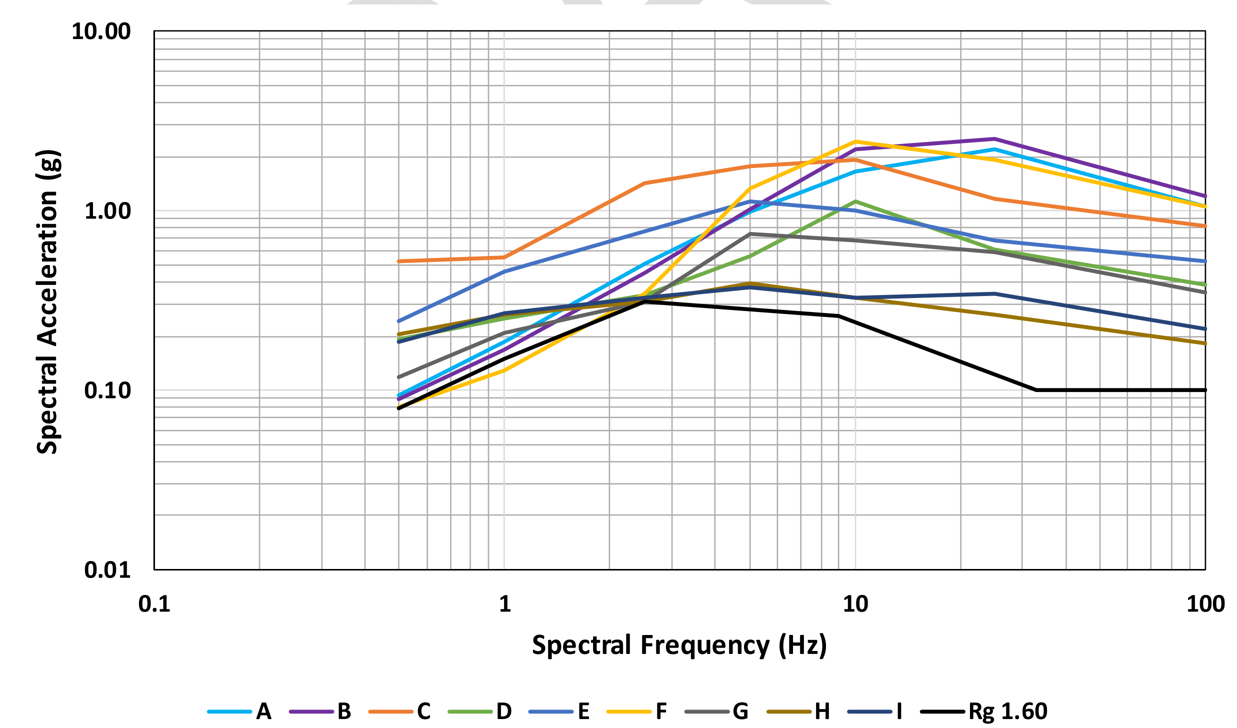

Figure 3-6 UHRS for nine CEUS sites corresponding the 1×10^{-5} exceedance frequency

ASCE 43 has minimum PGA values smaller than 0.1 g because SDC categories in ASCE 43 were developed in light of DOE facilities which include non-reactor nuclear facilities that may not require the same levels of seismic design as NPPs. Because only NPP designs are evaluated, the discussion with respect to 0.1 g PGA minimum is included in the following discussion.

Using the above UHRS spectra, SSDRS[5] associated with SDC–3, SDC–4, and SDC–5 are computed for three site conditions: (1) hard rock; (2) soil over rock; and (3) deep soil. Figure 3-7 shows DRS for the rock site. Figure 3-8 shows results for a selected case of a soil over rock condition, and Figure 3-9 shows results for a selected deep soil site.

As is typical at many CEUS sites, site-specific PSHA results indicate stronger ground motions at higher spectral frequencies and weaker ground motions at lower frequencies when compared to the RG 1.60 spectra anchored at PGA. The RG 1.60 spectrum was first developed in the 1970s and was based on a limited set of earthquake records from California. For a rock site (Figure 3-7), the RG 1.60 ground motion exceeds the SSDRS-5 motions for spectral frequencies of 2 Hz and lower. At 10 Hz, the RG 1.60 ground motion is considerably stronger than the SSDRS-3 ground motion at frequencies below 10 Hz. For sites with the soil over rock site (Figure 3-8), the differences are less pronounced. For the deep soil site (Figure 3-9), the SSDRS ground motions are, in general, weaker than the RG 1.60 spectrum. However, it is noted that the deep soil site represents a site near the Gulf of Mexico, in a region of infrequent seismicity. The implications of the differences among the SSDRS ground motions and the RG 1.60 spectra anchored at PGA bear on aspects of site-specific design activities and decisions, and they may also be important in the context of a certified design, which needs to be feasible for a variety of site conditions.

Ground motions at the nine CEUS sites (PGA and 5 Hz) for SSDRS-3, SSDRS-4, and SSDRS-5 are provided in Table 3-4. The ratios of the SSDRS-3 and SSDRS-4 ground motions to the SSDRS-5 ground motions are given in Table 3-5. The ratios of SSDRS-3, SSDRS-4, to SSDRS-5 illustrate the potential reduction in seismic demand if a lower SDC is used in the design of an SSC, assuming the design maintains a consistent LS for all SDCs. As discussed in Chapter 4, the reduction in seismic demand, by itself, may not control the design of an SSC because of many other complex design factors. Also note that the performance target for exceeding the LS is different in each of the SDC categories.

As evident from the results in Table 3-5, the PGA and 5 Hz values are 30 to 50% lower for SDC–4 ground motions when compared to the SDC–5 levels. SDC–3 PGA and 5 Hz values are 50 to 70% lower compared to the SDC–5 levels. As the current design approach uses LS-D, the above reductions also provide insights into how the fragilities (i.e., median capacity) could change if the designs were anchored to categories other than SDC–5, but for the same LS-D (with the different performance targets). This is an important insight in that the relationship between the capacity of an SSC and the risk significance of that component can be demonstrated explicitly.

[5]The procedure to compute SSDRS is described in Section 4.2.2

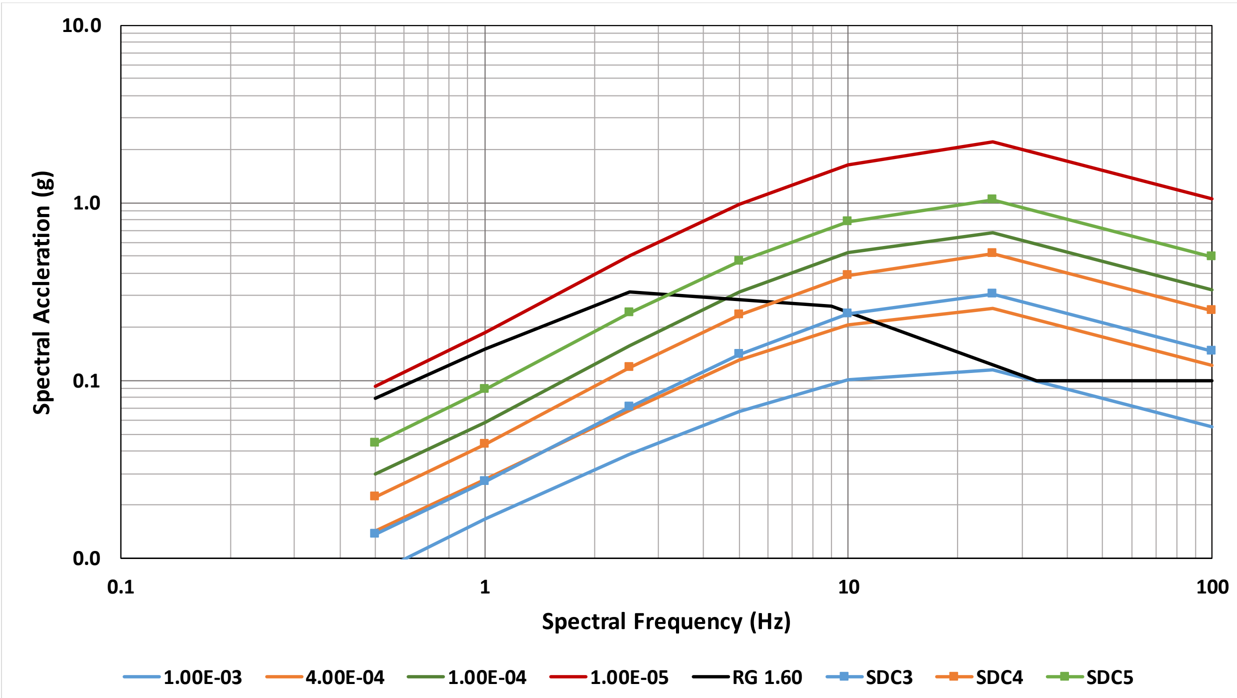

Figure 3-7 SSDRS for three SDC categories and four UHRS for Site A, compared to an RG-1.60 spectrum anchored at 0.1 g

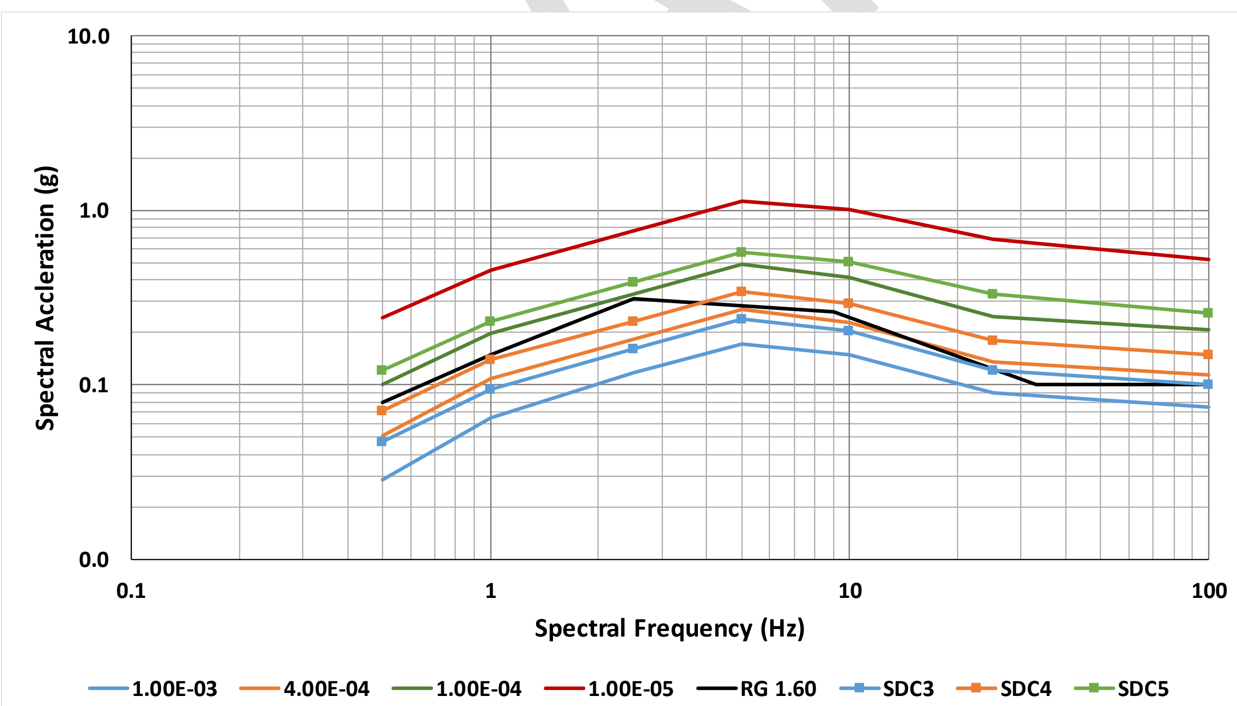

Figure 3-8 SSDRS for three SDC categories and four UHRS for Site E, compared to an RG-1.60 spectrum anchored at 0.1 g

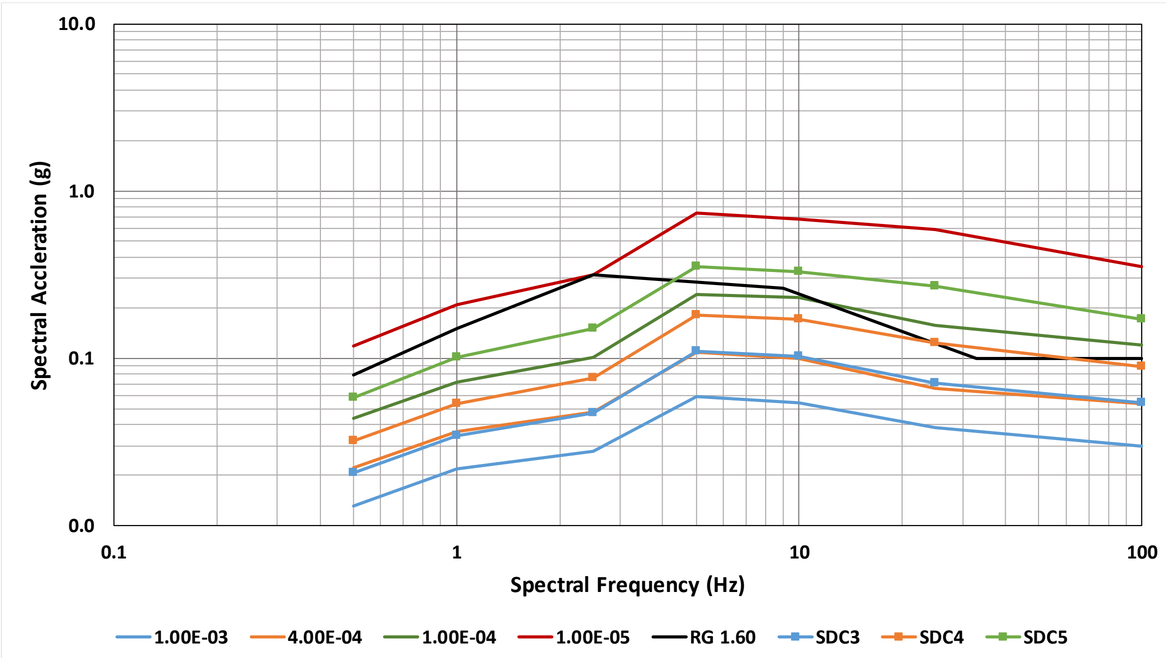

Figure 3-9 SSDRS for three SDC categories and four UHRS for Site G, compared to an RG-1.60 spectrum anchored at 0.1 g

Table 3-4 PGA and 5 Hz spectral acceleration ground motions for SDC–3, SDC–4, and SDC–5 at the Nine CEUS sites

Site	ASCE 43 DRS -PGA (g)			ASCE 43 DRS - 5 Hz (g)		
	SSDRS-5	SSDRS-4	SSDRS-3	SSDRS-5	SSDRS-4	SSDRS-3
A	0.50	0.25	0.15	0.47	0.23	0.14
B	0.57	0.27	0.17	0.49	0.24	0.15
C	0.44	0.29	0.21	0.92	0.60	0.42
D	0.19	0.11	0.07	0.28	0.16	0.10
E	0.26	0.15	0.10	0.57	0.34	0.24
F	0.51	0.25	0.15	0.61	0.28	0.16
G	0.17	0.09	0.05	0.35	0.18	0.11
H	0.09	0.05	0.03	0.20	0.12	0.08
I	0.11	0.06	0.04	0.19	0.11	0.08

Table 3-5 Ratios of SSDRS-3 and SSDRS-4 ground motions to the SSDRS-5 ground motions for all nine CEUS sites

Site	Ratios of PGA Values		Ratios of 5 Hz Values	
	SSDRS-4 /SSDRS-5	SSDRS-3 /SSDRS-5	SSDRS-4 /SSDRS-5	SSDRS-3 /SSDRS-5
A	0.49	0.29	0.50	0.30
B	0.48	0.30	0.50	0.30
C	0.67	0.49	0.65	0.46
D	0.56	0.37	0.57	0.37
E	0.57	0.39	0.60	0.42
F	0.50	0.30	0.45	0.26
G	0.52	0.32	0.51	0.31
H	0.55	0.38	0.58	0.40
I	0.58	0.40	0.60	0.42

Figure 3-10 through Figure 3-12 plot the SSDRS-5, SSDRS-4, and SSDRS-3 for all 9 sites. These plots also include CSDRS-5, CSDRS-4, and CSDRS-3 curves, which envelop the SSDRS-5, SSDRS-4, and SSDRS-3 curves, respectively. These CSDRS curves represent examples of design certification motions for the three SDC categories. In current practice, and in order to be generic to all sites, the CSDRS ground motion spectrum is developed to envelop ground motions at all 69 CEUS sites. While it is very likely that the same approach will be used for the advanced reactors to select the CSDRS spectrum, one crucial difference under the enhanced RIPB concept proposed in this report, is that there may more than one CSDRS, depending on how many SDC categories are selected and used for various SSCs in the seismic design.

Because of the enveloping of multiple site conditions, the CSDRS for all SDC categories will exceed the NRC's current minimum ground motion requirement of 0.10 g PGA. The annual exceedance frequencies of the CSDRS ground motions are lower than the corresponding annual exceedance frequencies of the underlying SSDRS motions.

As shown in Figure 3-13 and Table 3-6, the LS-D corresponds to essentially elastic response. Limit states C, B, and A permit progressively increasing permanent deformations and excursions into the inelastic regime. Because of the additional energy losses due to inelastic behavior, a component or a structural element is subjected to lower seismic demands compared to the elastic demand for a same design basis ground motion. Table 3-7 shows how, according to ASCE 43, forces for a shear controlled reinforced concrete shear wall are reduced for LS-A, LS-B, and LS-C compared to the LS-D forces. Chapter 4 includes a discussion of a shear wall design for different SDC and LS combinations to better assess the design process and implications.

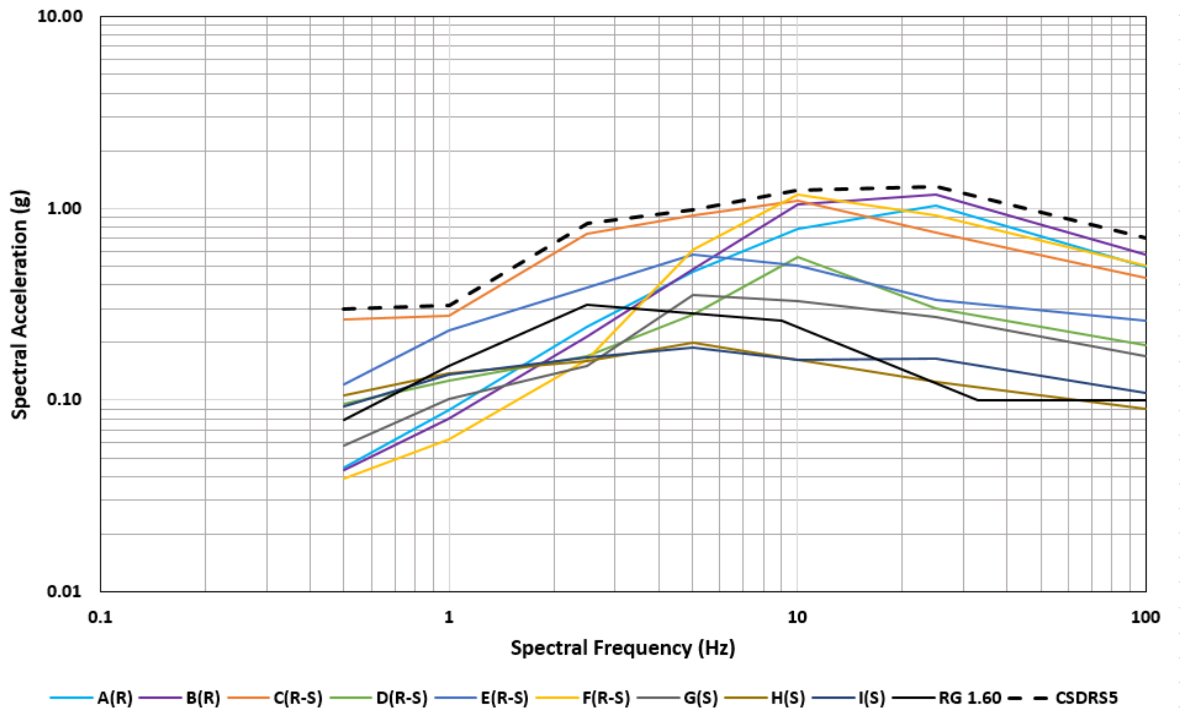

Figure 3-10 SSDRS-5 for all nine CEUS sites compared an RG-1.60 spectrum anchored at 0.1 g and the corresponding CSDRS

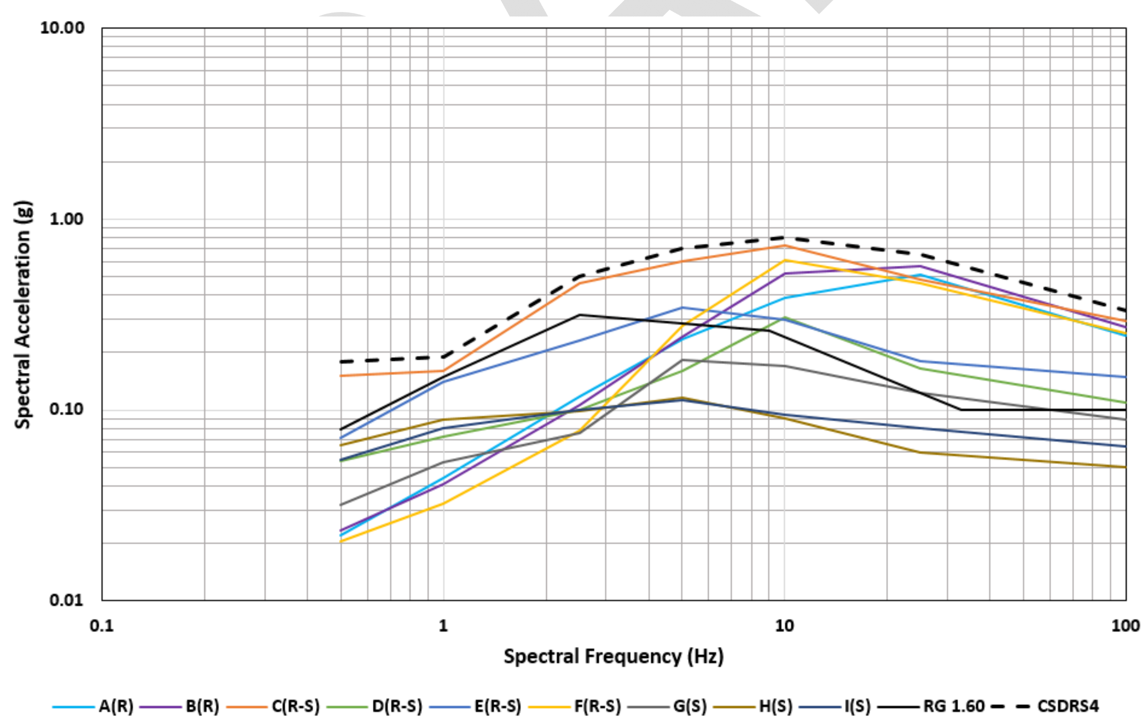

Figure 3-11 SSDRS-4 for all nine CEUS sites compared an RG-1.60 spectrum anchored at 0.1 g and the corresponding CSDRS

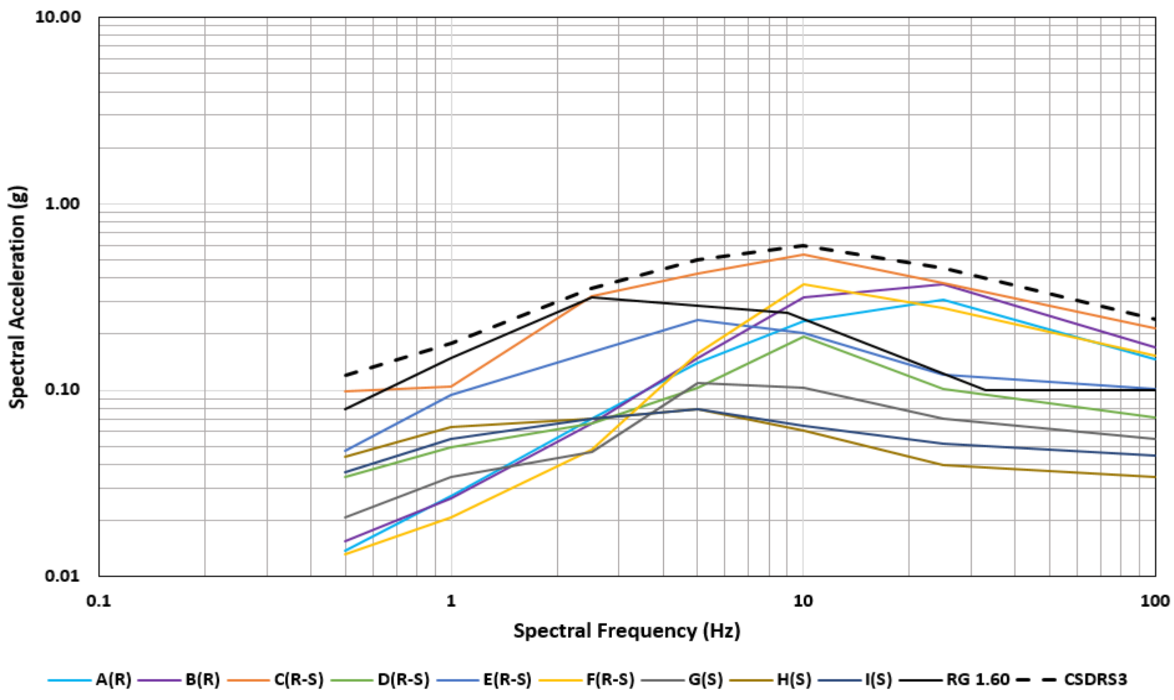

Figure 3-12 SSDRS-3 for all nine CEUS sites compared an RG-1.60 spectrum anchored at 0.1 g and the corresponding CSDRS

Table 3-6 Deformation and damage by limit state

Limit State	Expected Deformation	Expected Damage
LS-A	Large permanent distortion, short of collapse	Significant damage
LS-B	Moderate permanent distortion	Generally reparable
LS-C	Limited permanent distortion	Minimal damage
LS-D	Essentially elastic behavior	Negligible damage

Table 3-7 Reduction in seismic demand for a shear wall due to inelastic deformation

Reinforced concrete shear walls, in-plane	Ratio of forces for different limit states		
Shear controlled walls	LS-A/LS-D	LS-B/LS-D	LS-C/LS-D
Aspect Ratio: height/length < 2.0	0.50	0.57	0.67

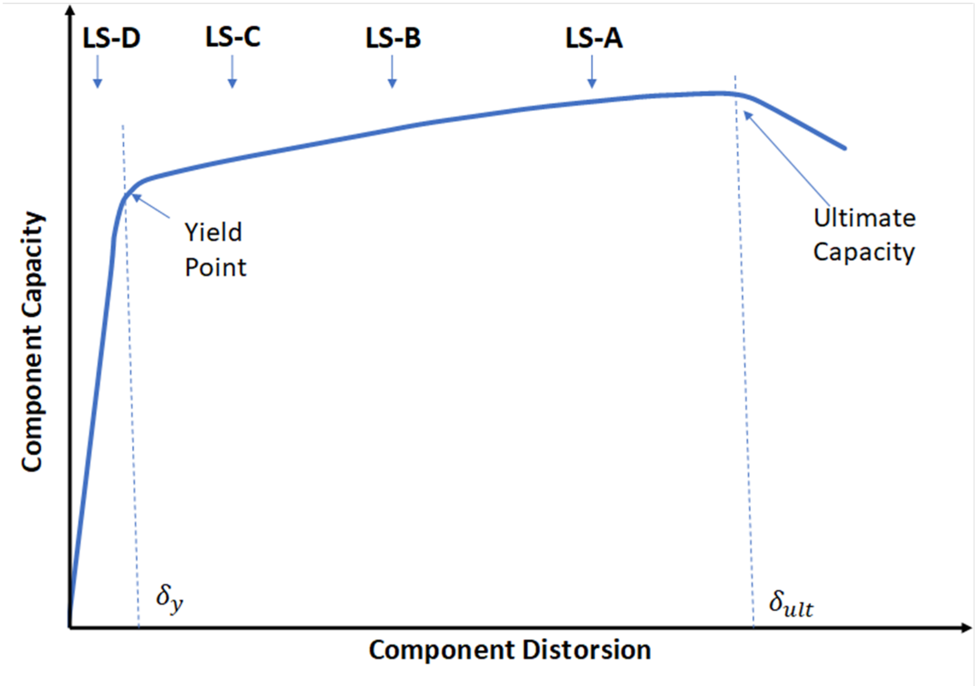

Figure 3-13 Typical load-deformation curve and limit states (adapted from ASCE 43)

3.3.1.2 Considerations Related to Selection of Limit State

In the enhancements to seismic design proposed in the report, the selection of the appropriate design limit state is also a fundamental decision in addition to the selection of the SDC. The damage limit states are shown in Table 3-6, per ANS Standard 2.26 (ANSI/ANS, 2004). Within the current ANS 2.26 standard, LS-D is the only limit state applicable to NPP design. The other less stringent limit states are for nuclear facilities with lower risk profiles. Examples in Chapter 4 of this report considered less stringent limit states for designing SSCs in order to evaluate how this aspect of NPP design contributes to risk.

Figure 3-13 illustrates the definitions of Table 3-6 in a graphical form and provides additional insights, where δ_y is the distortion or deformation at yield, δ_{ult} is the distortion or deformation at failure, and LS is the limit state from Table 3-6. A comparison of values in Table 3-5 and Table 3-7 provides insights into the seismic demand reductions based on a choice of SDC versus a choice of a LS. The shear wall example in Chapter 4 examines a few combinations of SDC categories and limit states. The selection of LS-B and LS-A is not a likely option for safety related components (rows 1 to 4 of Table 3-2) for reactor and spent fuel pool risks; these limit states may be appropriate for other radiological sources, such as waste holdup tanks.

Another qualitative insight from Figure 3-13 is related to the available margin of an SSC to ultimate failure (e.g., fragility failure mode when used in a SPRA), for beyond design basis loads. As the inelastic deformations are permitted for LS-C, LS-B, and LS-A, the margins to ultimate failure deformation are reduced. Fragility calculations should reflect these reductions in capacities.

3.3.1.3 Other Considerations Related to Choices of SDC Categories and Limit States

The discussion in the sub-sections 3.3.1.1 and 3.3.1.2 was primarily concerned with the potential benefits of selecting alternative SDC categories and limit states in reference to the current practice of using SDC–5 and LS-D for all safety related SSCs in NPPs. These benefits arise are primarily from reduced seismic design demands that could lead to more uniform margins and conservatisms consistent with the risk importance, so long as the design meets the overall risk metrics. However, there are additional practical and regulatory considerations that need to be balanced against potential benefits in the selection of less stringent SDCs and LSs.

One important consideration is the stability of the design, especially at the design certification stage. The seismic design at this stage should result in a power plant that is viable for a range of seismic conditions without requiring substantial site-specific modifications. Stability also includes operability considerations over the lifetime of a plant, especially as new knowledge about the seismic hazard emerges. The availability of design details at the time decisions are made about SDC categories and LS is also an important factor. The lack of design detail may require relatively conservative choices of the SDC and LS in order to avoid excessive design iterations and significant design changes. Decisions and choices at the design certification stage should be able to identify clear site-specific interfaces and activities that can be easily implemented. In recent experience with some Combined Operating License (COL) applications, seismic interfaces have emerged as an important issue. The requirements related to the minimum design ground motion are particularly important for site-specific component design and selection of an SDC for the site-specific ground motion.

Issues related to the designation of the operating basis earthquake also should be considered. Appendix S to 10 CFR Part 50 requires:

> *"(i) The Operating Basis Earthquake Ground Motion must be characterized by response spectra. The value of the Operating Basis Earthquake Ground Motion must be set to one of the following choices:*
>
> *(A) One-third or less of the Safe Shutdown Earthquake Ground Motion design response spectra. The requirements associated with this Operating Basis Earthquake Ground Motion in Paragraph (a)(2)(i)(B) (I) can be satisfied without the applicant performing explicit response or design analyses, or*
>
> *(B) A value greater than one-third of the Safe Shutdown Earthquake Ground Motion design response spectra. Analysis and design must be performed to demonstrate that the requirements associated with this Operating Basis Earthquake Ground Motion in Paragraph (a)(2)(i)(B)(I) are satisfied. The design must take into account soil-structure interaction effects and the duration of vibratory ground motion."*

Consistent with the recent design certification applications, option A seems most likely. However, the selection of multiple SDC categories could result in multiple operating basis earthquake (OBE) ground motions, complicating decisions about plant shutdown and restart should an earthquake occur. This situation does exist currently to a limited extent in that the ISG-01(NRC, 2008) discusses the interpretations of the OBE for the certified design portion of a plant and the site-specific designed portion of a plant. The more important consideration with respect to the OBE is restart after an earthquake, especially the ability or need to restore the pant to its original licensed conditions, should the design allow limited damage, consistent with LD-C for example. In the current practice of designing to the LS-D, the elastic response (no permanent damage) means that aftershocks are not an important factor in seismic design. One of the other factors is related to the combination of accident and earthquake loads. This question should also be considered in the selection of SDC categories and limit states.

Although many SDC and limit state combinations can be considered (16 in the ASCE 43 code, to be precise), practical applications should consider options of SDC–5 and LS-D, SDC–5 and LS-C, SDC–4 and LS-D, and SDC–4 and LS-C for safety related components related to the reactor and spent fuel risk. The SDC–3 and LS-D options may also be acceptable for a few SSCs. One important advantage of using LS-D is that the seismic responses and demands can be scaled from one SDC analysis to other SDCs, with some approximations, because of the linear responses. More detailed insights will be available on scaling approaches after completing some of the examples and analyses in potential future activities employing the methods discussed in Chapter 4.

3.3.2 Step 2—Seismic Design

Step 2 is to implement seismic design according to ASCE 43, ASCE 4, ASCE 1, applicable NRC and industry guidance, and codes and standards according to selected SDCs and LSs. Current design approaches and the dominant experience is based on SDC–5 and LS-D approaches of the ASCE 43, 4, and 1, codes. Thus, maintaining LS-D while relaxing the SDC requirement is a reduction in the design ground motions and does not affect the response analysis or design methods (except for some changes in numerical values of parameters, such as damping). Although approaches are outlined in ASCE 43 and ASCE 4 for designs that include LS-C and lower, experience in applying these inelastic design options to nuclear grade structures and equipment is limited. The examples discussed in Chapter 4 provide some insights into how combinations of alternative SDCs and LSs in NPP design might be realized within a RIPB framework.

During the iterative process, this step does not imply a rigorous re-design of the entire plant but allows as an option a design assessment of the components that are candidates for alternative SDC and LS designations, so that more realistic fragilities can be estimated in the next step.

3.3.3 Step 3—Fragility Determination

In Step 3 the fragilities of the SSCs included in the SPRA model are determined in consultation with the LMP Component Group and in accordance with the applicable ASME/ANS SPRA standard. In developing these fragilities, consideration of some important aspects of the SPRA beyond an evaluation of risk and performance metrics are necessary. These considerations include the SPRA's role in supporting safety and seismic design classifications of SSCs that are needed for NRC licensing.

Available details of designs dictate to a large extent the realistic and component specific fragilities. Based on current experience[6], it is unlikely that complete realistic fragilities will be available or developed at the initial design stage; however, the goal of the seven-step approach is to be as realistic as possible because the basic design parameters for SSCs are derived through an iterative process. Generic fragilities currently being used in the design of NPPs are based on LS-D. Also, the designs of operating reactors include other factors, such as the

[6]Based on the experience with design certification reviews, the most optimistic expectation was to achieve conservative fragilities for structures while assigning the fragility values for component and equipment deferring their confirmations at COL (or before fuel load).

combination of accident and seismic loads, design for other external hazards, and a more conservative design basis in some cases. It is not clear that such margins will exist in future NPP designs.

In the context of the proposed seven-step design process, the separation of variables (SOV) method based on the currently available industry guidance should be used in order to develop more realistic fragilities, as the outcome of the process will define the design level for a component through an iterative process. Additionally, the screening of SSCs should be limited in order to gain the maximum benefit from the reclassification of an SSC to different seismic design levels according to risk criteria.

Finally, current experience in developing fragilities is related to components designed to LS-D. In the seven-step proposed process, it will be necessary to develop fragilities for components designed to different limit states and different damage levels. Chapter 4 discusses the approaches for how a component fragility associated with LS-D may be modified to reflect the application of LS-C.

3.3.4 Step 4—Perform a Seismic PRA

In Step 4, the analyst performs a SPRA using the probabilistic seismic hazard curves and the fragilities developed in Step 3 and the SPRA models developed in accordance with the applicable ASME/ANS codes. One other crucial difference between the current licensing procedure and this proposed seven-step seismic design process is the role of a PRA. In current licensing procedures, PRAs are required at various stages (i.e., design certification, COL and fuel loading), but these PRAs are not included in the licensing basis. In the proposed seven-step process, the PRAs play a more significant role in determining licensing basis events and other considerations and become a part of the licensing basis.

Performing a SPRA at the design certification stage is also a departure from the current approach of doing a PRA-based margin analysis (ISG-020, NRC, 2010a). The current PRA-based margin approach does not require a seismic hazard analysis. In this seven-step process, seismic hazard curves are required at the design certification stage. Although several approaches can be considered, two approaches are described here in order to illustrate how seismic hazard information can be incorporated into the design certification process.

There are three aspects of the PSHA results that are important to present the two approaches. The first is the hazard intensity at the annual exceedance frequency corresponding to the selected SDC (e.g.,1×10^{-4}). The second is the slope of the hazard curve over a range of annual exceedance frequencies (e.g., 1×10^{-2} to 1×10^{-7}) that can be used to evaluate the beyond design basis ground motions. Given the same component designed to the same design basis ground motion, a steep hazard curve over this range will result in smaller failure probabilities than the failure probabilities from shallow hazard curve slopes. The third is the spectral shapes of the response spectra (UHRS or GMRS), which are site-specific because they depend on the nature of the controlling seismic sources, ground motion attenuation, and the site geotechnical conditions (hard rock, still soil, or soft soil). As shown in Figure 3-3 through Figure 3-6, the site-specific UHRS spectra differ significantly from site to site. These site-specific differences in the response spectral shape and associated fragility curves need to be accounted for in the generic design certification process. In addition, it will be very instructive if some site-specific issues can be examined during the design certification stage to assure more robust design and identify the necessary interfaces for a combined license application.

Approach one to incorporate seismic hazard information into the design certification process regards the use a bounding hazard curve that envelops site-specific hazard estimates at the sites where the design may be located. With the availability of probabilistic seismic hazards analysis for all CEUS sites, this can be accomplished in straightforward manner. Figure 3-2 showed hazard curves for nine sites with varying site conditions. However, in this approach one the issue of different UHRS spectral shapes is not considered explicitly and may mask some important insights.

Approach two is more involved requiring additional fragility evaluations that account for various site conditions. In some ways, this approach is similar to the approach currently used to demonstrate the adequacy of a certified seismic design for multiple site conditions. In the certified design, various sites are selected, and site-specific soil structure interaction analyses are performed to determine site-specific seismic demands using the CSDRS. Results from all site cases are enveloped. Similarly, for conducting an SPRA, an applicant may choose some representative sites[7] and use site-specific hazard curves and site-specific UHRS shapes to demonstrate the acceptability of the design and compliance with risk and performance criteria. It is anticipated that the SPRAs would include non-seismic failures and human error probabilities (HEPs). As discussed in Step 5, this may be important input for the plant design.

3.3.5 Step 5—Check the proposed classification against the risk, defense-in-depth, reliability, and other qualitative criteria

In Step 5 the LMP Component Group evaluates the results to make sure that the individual event sequence results meet the F-C Target dose limits and the integrated risk criteria and identifies the risk significant LBEs that are within 1% margin close the F-C curves. This group also evaluates defense-in-depth adequacy, other qualitative factors related to the risk-informed decision-making (e.g., balance between prevention and mitigation, avoidance of singleton failures that control the risk), and other LMP guidelines. Based on this evaluation, the LMP Component Group determines whether assignment of an SSC to a different SDC is necessary or needs to be strengthened. This group may also identify opportunities (in consultation with the seismic design engineers) to design an SSC to a lower SDC or a less stringent LS.

In some of the current LWR SPRAs, non-seismic failures and HEPs have been found to be dominant or important contributors. In such cases, the required design changes to assure that the design meets the risk criteria may not be related to the seismic design. It is important that the SPRA models include these failure modes to the extent possible.

In this step, feedback is provided to the seismic design and SPRA teams to reclassify the components as necessary and to then recompute the fragilities. The examples discussed in Chapter 4 provide additional insights into this step.

The end states of the event sequences for the advanced reactors SPRAs (or PRAs for any other initiators) can lead to much different end states compared to the end states of a Level 1 LWR PRA, that are CDF and LERF. The examples discussed in Chapter 4, when carried out in potential future activities, may provide insights into the important question of how external hazard event sequences can be assessed to evaluate compliance with the F-C Target.

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[7] Based on discussions with its developers, the approach two appears to be consistent with the LMP initiative.

Because the risk criteria for the advanced reactors are used for design purposes and that SPRAs are performed at the design certification stage, the current criteria of requiring high confidence - low probability of failure at 1.67 the design basis earthquake is neither applicable nor necessary. However, for an LWR design, retaining this concept may be useful.

3.3.6 Step 6—Iteration

Steps 2 through 5 are repeated until the process is stabilized and all risk criteria are met. There may be opportunities for streamlining this iterative process by using some simple approaches to adjust fragilities for different SDC and LS combinations.

3.3.7 Step 7—Final SSC Categorization for Seismic Design

The final SSC categorization becomes the basis for the licensing and design of the plant. The interface requirements for a COL application are based on this seismic design categorization and the final design is implemented using it. This final SSC categorization may be different than assumed in Step 1. For example, as shown in Table 3-8, some of the safety-related components could be designed to a lower SDC but retain their status as safety related. However, some NSRST components will perhaps need to be strengthened using a higher SDC, but with no special treatment. The final categorizations in Table 3-8 are for illustration purposes. In this example, there are four licensing and design basis ground motions: CSDRD-5, CSDRS-4, SSDRS-5, and SSDRS-4. An appreciation of the effort involved in reaching this final step will become clear after performing the feasibility and demonstrations identified in Section 4.3.

Table 3-8 Example of final SDC and LS following the seven-step seismic design process

Design Categories	Initial ASCE SDC and LS	Final ASCE SDC and LS
SSCs selected by the designer to perform required safety functions to mitigate the consequences of DBEs to within the F-C target, and to mitigate DBAs to meet the dose limits of 10 CFR 50.34 using conservative assumptions	SDC–5 LS D CSDRS-5 (or SSDRS-5)	SDC–5 LS D CSDRS-5 (or SSDRS-5)
SSCs selected by the designer to perform required safety functions to prevent the frequency of BDBEs with consequences greater than 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C target.	SDC–5 LS D CSDRS-5 (or SSDRS-5)	SDC–5 LS C CSDRS-5 (or SSDRS-5)
Non-safety related SSCs relied on to perform risk significant functions. Risk significant SSCs are those that perform functions that keep LBEs from exceeding the F-C target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs.	SDC–5 LS D CSDRS-5 (or SSDRS-5)	SDC–4 LS D CSDRS-4 (or SSDRS-4)
Non-safety related SSCs relied on to perform functions requiring special treatment for DID adequacy.	SDC–5 LS D CSDRS-5 (or SSDRS-5)	SDC–4 LS C CSDRS-4 (SSDRS-4)

3.4 Application to ANLWRs or LWRs

Although the LMP framework is for advanced non- light-water reactors, the process of Sections 3.2 and 3.3 is also applicable to ALWRs or LWRs, if adjusting for different risk criteria. The feasibility of the process may be demonstrated to some extent by using SPRAs of existing LWRs.

There is extensive experience with the seismic designs of LWR SSCs, and many SPRAs are currently available with more rigorous and plant-specific fragilities. More importantly, some of these SPRAs have been used to obtain certain regulatory relief using the flexibility allowed in 10 CFR 50.69, Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors. In the 50.69 SSC Categorization Process, SSCs must be categorized as Risk-Informed Safety Class (RISC), RISC–1, RISC–2, RISC–3, or RISC–4

SSCs using a process that determines whether an SSC performs one or more safety significant functions and identifies those functions. These risk categories are briefly defined as

- RISC–1 SSCs: safety-related SSCs that perform safety significant functions
- RISC–2 SSCs: non-safety-related SSCs that perform safety significant functions
- RISC–3 SSCs: safety-related SSCs that perform low safety significant functions
- RISC–4 SSCs: non-safety-related SSCs that perform low safety significant functions

In the context of these definitions, a safety significant function means a function whose degradation or loss could result in a significant adverse effect on defense-in-depth, safety margin, or risk.

Figure 3-14 shows these risk categories in a graphical form. Because of the applications of 10 CFR 50.69 to several operating plants, reliable information now exists regarding how the SSCs are distributed among different risk categories. Availability of this information, detailed SPRA models, and fragility calculations affects how some of the seven-steps would be implemented.

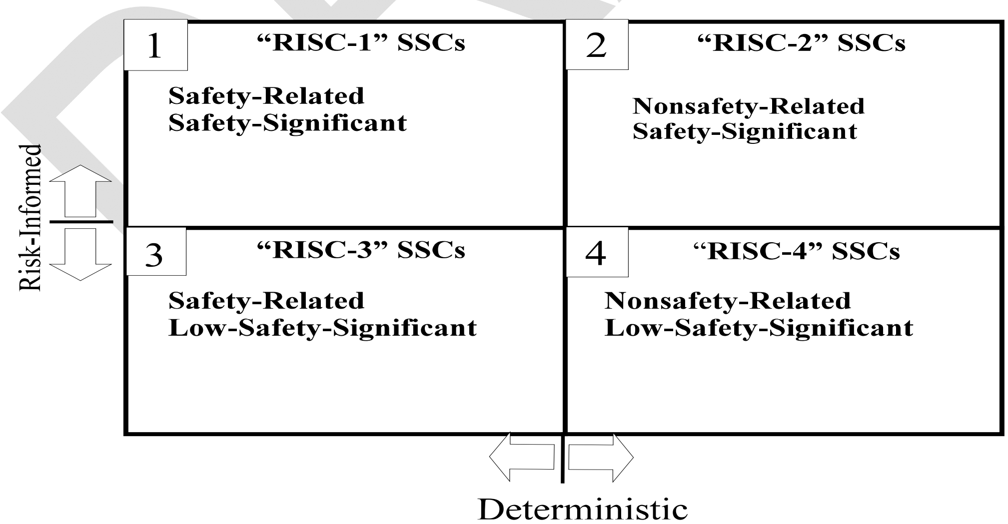

Figure 3-14 The 50.69 RISC matrix

Step 1: Initial Selection of the ASCE 43 SDC and LS categories

Because of the availability of several 50.69 analyses, this initial selection is much better informed based on this existing information. The original selection of SDCs and LSs can be made using the RISC categories in Figure 3-14. For example, for SSCs belonging to RISC-1, SDC–5 and LS-D may be chosen. For SSCs belonging to RISC-2, SDC–5 and LS-C (or SDC–4 and LS-D) may be chosen.

Step 2: Seismic Design

Step 2 in Section 3.3.2 also applies for LWRs. However, the extensive experience with these types of reactors should make it possible to have more detailed information.

Step 3: Fragility Determination

Step 3 in Section 3.3.3 also applies to some extent for light water reactors. However, the extensive experience with these types of reactors should make it possible to have more detailed information.

Step 4: Perform Seismic PRA

As described in Section 3.3.4, it is important to develop an SPRA using the fragilities developed in Step 3 and the SPRA models developed in accordance with the applicable codes, and those used in developing LBEs. The considerations discussed earlier regarding the selection of probabilistic seismic hazards are also applicable. Although, the current approach for ALWRs permits the performance of a PRA-based margin approach for the design certification stage, for this process, an SPRA needs to be performed in the context of establishing LBEs and the basis for the seismic design of SSCs.

After completing the SPRA, results are provided to the LMP Component Group to check whether the results meet the applicable risk performance targets, such as CDF and LERF and other qualitative factors.

Step 5: Check the proposed classification against the risk, defense-in-depth, and other qualitative criteria

The LMP Component Group evaluates the results to make sure that the individual event sequence results are within the risk target (assuming the SPRA outputs dose estimates and frequencies of event sequences), the integrated risk criteria are met, and the risk significant LBEs are within applicable risk criteria. Based on the above analysis, the LMP Component Group determines whether a reclassification of an SSC is necessary, or some other actions need to be taken. This group may also identify opportunities to recommend that a component be designed to a lower SDC or a less stringent LS. There could also be non-seismic related factors, such as HEPs, that may be controlling factors and need to be addressed. This feedback is provided to the seismic design and SPRA teams so that they can reclassify the SDC categories and LS of the components as necessary, and then recompute their fragilities.

The current approach requiring demonstration of plant-level performance target of high-confidence, low probability failure (HCLPF) level of 1% at 1.67 times the design basis ground motion might still be useful if the end states of a PRA are similar to the CDF and LERF.

Step 6: Iteration

Step 6 in Section 3.3.6 also applies for LWRs.

Step 7: Final SSC Categorization for Seismic Design

Step 6 in Section 3.3.7 also applies for LWRs.

4 APPROACHES TO EVALUATE THE FEASIBILITY OF THE SEVEN-STEP SESISMIC DESIGN PROCESS

4.1 Background

In this chapter, the feasibility of the seven-step seismic design process described in Chapter 3 is evaluated. This chapter includes preliminary and exploratory examples, which are used to identify future potential activities to support the development of technical basis of the U.S. Nuclear Regulatory Commission (NRC) guidance on seismic safety design. These examples and the methods and procedures will also form a basis for discussions with the stakeholders to gain alignment and for potential collaborative activities.

There are no actual examples of a commercial nuclear power plant (NPP) design that utilized the American Society of Civil Engineers (ASCE) 43[1] (ASCE/SEI, 2005) performance targets, although seismic designs based on the most stringent Seismic Design Category (SDC) and Limit State (LS), SDC–5 and LS-D, are quite similar to current design practice. It is difficult to directly assess the potential impact on seismic risk evaluations should the full ASCE 43 approach for seismic design be implemented. Alternative fragilities of structures, systems, and components (SSCs) are needed to assess a more comprehensive implementation of ASCE 43.

As discussed in Chapter 3, an important difference between the currently operating plants [licensed prior to publication of revised Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100 in 1997] and the ASCE 43 approach for SDC–5 with LS-D is the design basis ground motion. The current plants were designed to a deterministic, broad-band response spectrum anchored to a peak ground acceleration and determined in accordance with the 10 CFR Part 100, Appendix A requirements. The ASCE design response spectra (DRS) for SDC–5 is a modified uniform hazard response spectra (UHRS) (with the annual frequency of exceedance less than 1×10^{-5}).

Three proposed approaches are described in this chapter that could be used to evaluate the feasibility of the seven-step process defined in Chapter 3, and to gain insights on the scope of its potential benefits and associated implementation costs.

<u>Example 1</u>: Generic fragility calculations are performed for selected combinations of SDC and LS categories using the underlying assumptions of ASCE 43 and 4-16 (ASCE/SEI, 2017) codes with respect to performance goals.

<u>Example 2</u>: A simple structural element is designed using selected combinations of SDC and LS categories and using the ASCE 43 and 4 codes, and the same element is also designed using the conventional approach in the Standard Review Plan (SRP). Fragilities are developed for each case and compared and are then used to compute failure probabilities.

<u>Example 3</u>: This example is a multi-layered example examining the application of the seven-step process for light-water reactors (LWRs) and for advanced reactors, with the goal to identify potential activities that would strengthen the development of technical basis for future

[1]As noted in Chapter 3, the currently published ASCE Standard 43 is Version 43-05, published in 2005. A new version is planned to be published soon. The basic philosophy of both versions is same. For the purposes of this Chapter, ASCE 43 is cited in the text without the year of publication.

NRC guidance on seismic safety. These activities could involve a combination of simplified probabilistic risk assessment (PRA) models and complex modeling over a longer term. These activities are recommended not only to evaluate risk impacts of the proposed process, but to better assess implementation issues. This includes developing detailed ground rules, understanding the efforts involved, and providing guidance on the key managerial and technical decisions.

The Example 3 draws information from NUREG/CR-7214, "Toward a More Risk-Informed and Performance-Based Framework for the Regulation of the Seismic Safety of Nuclear Power Plants" (Budnitz and Mieler, 2016). This report set forth and evaluated attributes of an ideal risk-informed and performance-based framework for assuring seismic safety.

The Example 3 also examines whether there are any methods that would allow an analyst to recompute existing fragilities of SSCs as if they were designed to the various ASCE 43 SDC and LS combinations. If such is possible, it may be feasible to use existing SPRAs to more fully evaluate the risk impact of different SDC and LS categories by using an approach similar to the approach examined in NUREG/CR-7214.

4.2 Example 1—Computation of Generic Fragilities

This section describes how the fragilities of SSCs that are designed to the ASCE 43 criteria could be derived. For this approach, the SSC are assumed to be designed to the full limits of design criteria. In practice, there will be design margins which may vary from SSC to SSC. However, these design margins for SSCs are not relied on for fragility evaluations at the conceptual design stage. Seismic loading is typically the dominant load, but the design could also be controlled by other loads and load combinations.

For the sake of simplicity, the SSC fragility is derived in terms of median peak ground acceleration capacity and the composite variability. Other measures such as the spectral acceleration at a specified frequency could also be used. The generic estimates of safety factors and their variabilities are used for this stage of the fragility calculation. The fragility estimates could be modified at later stages using site- and plant-specific data.

Normally, fragility evaluation of an SSC accounts for the actual design data. For example, for a shear wall, the design data includes wall thickness, reinforcement, nominal concrete strength, and the earthquake-imposed load. For a switchgear, these design data include dimensions, anchorage, qualification test results, and in-structure response spectra. If the response analysis has been performed to the Design Response Spectra (DRS) input, the critical shear wall among all the walls at a particular floor will be identified for fragility assessment. Similarly, the switchgear mounted on different floors may have different designs and anchorage and the fragility assessment would account for these differences.

Because this level of data will not be available at the conceptual design stage, the objective at this stage is to demonstrate how the fragilities of SSCs are developed if they just meet the ASCE 43 criteria. Therefore, the shear wall is assumed to barely meet the ASCE 43 design criteria [such as DRS, seismic response analysis, ACI 349 standard (ACI, 2013) requirements etc.]. Similarly, the electrical equipment is tested to Institute of Electrical and Electronics Engineers (IEEE) requirements for the specified in-structure response spectra (ISRS) based on ASCE 4 seismic response analysis requirements.

These fragility estimates would be used in the PRA (at the conceptual design stage) for all similar SSCs. By this simplification, however, the important feature of the variation of seismic response due to differences in physical location of SSCs (buildings and floors) is not captured. The fragility analysis can be refined as more design details and information become available in subsequent phases of the project.

4.2.1 ASCE 43 Design Criteria

Seismic design of SSCs according to ASCE 43 could be summarized by the following steps:

1. Assume SDC–5 for all safety related SSCs.

2. Select performance goal (P_F) = 1×10^{-5} per year for SDC–5.

3. Develop design basis earthquake (DBE) DRS[2] using the equation DRS = SF × UHRS for the P_F, where SF is the scale factor and UHRS is the uniform hazard response spectrum at exceedance frequency $H_P = P_F$.

4. ASCE 43 specifies the following additional performance targets: 1% probability of unacceptable performance for DBE shaking and 10% probability of unacceptable performance at 1.5 DBE shaking. The design criteria in the selection of DBE, seismic response analysis per ASCE 4-16 for design codes such as ACI 349 for concrete structures and AISC for steel structures are shown to meet these performance targets.

5. Assume LDS-D; elastic behavior

6. Compute the scale factor (SF) for a spectral frequency specific to the site seismic hazard, which is given by the following equations:

   $SF = \max(SF_1, SF_2, SF_3)$

   $SF_1 = A_R^{-1}$

   $SF_2 = 0.6\ A_R^{-0.2}$

   $SF_3 = 0.45$

   where, $A_R = SA_{HP}/SA_{HD}$. In this equation, SA_{HP} = spectral acceleration at the exceedance frequency H_P, and SA_{HD} = spectral acceleration at the exceedance frequency H_D. Note $H_D = 10\ H_P$ where $H_P = P_F$.

7. Based on Site A seismic hazard results for peak ground acceleration (PGA) (e.g., Figure 3-2) and 5 Hz (Table 3-3). The values for the parameters are given as:

   $SA_{HP} = 1.10$ g, $SA_{HD} = 0.36$ g

   $A_R = SA_{HP}/SA_{HD} = 2.77$

   $SF_1 = 0.36$

[2]The procedure used to develop the DRS is based on a newer (and unpublished) version of ASCE 43. For the purposes of this evaluation, the procedure in ASCE 43-05 produces the same DRS.

$SF_2 = 0.49$

$SF_3 = 0.45$

DBE DRS = 0.49 UHRS and DBE PGA = 0.50 g

8. Perform seismic response analysis following ASCE 4-16; it achieves an 80% probability of non-exceedance response given the DBE shaking.

9. Design structural elements (e.g., shear walls, beams, columns, tanks etc.) using ACI 349 and AISC codes.

10. For equipment qualified by testing, use the test response spectrum (TRS) as 1.33 times the required response spectrum (RRS); RRS at the equipment mounting (floor) level is obtained for the DBE DRS and seismic response analysis per ASCE 4-16.

4.2.2 Development of Fragilities

At the conceptual (or design certification) stage, all is known is that SSCs will be designed to meet the ASCE 43 design criteria. In practice, not all SSCs will be designed to the design limits; other loads and load combinations could govern the design dimensions. Generic variabilities (β values) are used in the following calculations.

4.2.3 Structural Fragility

A shear wall in a safety related building in the plant is assumed for this example. Following cases of structural fragility is evaluated for the shear wall.

4.2.3.1 Assess the fragility if the shear wall designed to Limit State D

The median ground acceleration capacity, A_m, can be written as:

$$A_m = F_T \times DBE \times PGA$$

where, the total factor, $F_T = F_{Strength}\ F_{\mu}\ F_R$

The strength factor, $F_{Strength,}$ is the product of the factor reflecting the uncertainty in the material property (reinforcing steel) (F_{mat}) and in the shear failure formula ($F_{formula}$). F_{μ} is the inelastic absorption factor, and F_R is the response factor. The numerical values of these factors are based on Electric Power Research Institute (EPRI) TR-103959 (EPRI, 1994) and past seismic PRAs. More recent test data reported in ASCE 43 could be used in future work.

$F_{mat} = 1.20$; $\beta c = 0.10$

$F_{formula} = 2.0$; $\beta c = 0.20$

$F_{\mu} = 1.80$; $\beta c = 0.20$

The response factor, F_R, is obtained by invoking the ASCE 4-16 goal of achieving the 80% probability of non-exceedance of response for DBE shaking

$F_R = \exp(0.842\ \beta_R)$ where $\beta_R = 0.35$

$F_R = 1.34$

Total factor of safety (F_T) = 1.20×2.00×1.80×1.34 = 5.80; $\beta_c = 0.43$

The median ground acceleration capacity, A_m, of the shear wall designed to LS-D is given by (5.8×0.50) 2.9 g.

High-confidence, low probability failure (HCLPF) capacity = **1.08 g**

4.2.3.2 Assess the fragility if the shear wall designed to Limit State C

If the shear wall is designed to LS-C, the design demand is reduced by a factor representing the inelastic energy absorption (see Eq. 5-1a of ASCE 43). All other factors being the same, the median ground acceleration capacity will also be reduced by this factor. Table 5-1 of ASCE 43 gives this reduction factor as 1.5 for LS-C.

Therefore, the median ground acceleration capacity of the shear wall designed to Limit State C is given by (2.9/1.5) 1.93 g.

HCLPF capacity = 1.08/1.5 = **0.72 g**.

Note by designing to LS-C, the shear wall will have less reinforcement (other design features such as span, height and wall thickness may not change). Designing for a lower LS would generally result in some cost savings.

4.2.3.3 Assess the fragility if the shear wall designed to SDC–4

If the shear wall is designed at SDC–4 for LS-D, the input to the seismic demand analysis will be based on the performance goal of 4×10^{-5} per year. The DBE PGA is 0.25 g, rather than the value of 0.50 g for SDC–5. The median ground acceleration capacity of the SDC–4 shear wall is given by (2.9×0.25/0.50) 1.45 g.

HCLPF capacity = **0.53 g**.

Note by designing the wall as SDC–4, the shear wall will have less reinforcement (other design features such as span, height and wall thickness may not change) because the DRS input is lower. Designing for a lower SDC would generally result in some cost savings

4.2.4 Equipment Fragility – Functional Failure

The RRS is calculated as the floor response spectra at the level of equipment mounting using the ASCE 4-16 procedure for the DBE shaking.

$F_R = \exp(0.842\ \beta_R)$ where $\beta_R = 0.35$

$F_R = 1.34$

TRS/RRS = 1.33

The qualification test capacity (TRS) is judged to be a 95% probability (confidence) value. Therefore, the median capacity factor $F_C = \exp(1.65\ \beta_C) = 1.39$ for $\beta_C = 0.20$

Total factor of safety = 1.34×1.33×1.39 = 2.48

Median ground acceleration capacity (A_m) = 2.48×0.50 = 1.24 g

Total variability in ground acceleration capacity (β_c) = $(0.35^2+0.20^2)^{1/2}$ = 0.40

HCLPF Capacity = **0.49 g**

4.2.5 Equipment Fragility–Anchorage Failure

Anchorage is designed such that the governing failure is ductile; therefore, the capacity of steel controls.

Tensile capacity of anchor = 1.6 F_a As, where F_a = 20 ksi and As = bolt area

Ultimate tensile capacity = ϕ F_{um} As; F_{um} = 58 ksi and ϕ = 0.9

Strength factor = code capacity / ultimate capacity = 1.80; β = 0.13

Designers typically use the equivalent static method for design of anchorage where the peak spectral acceleration is multiplied by 1.5 to calculate the anchor load. Therefore, there is a minimum of 1.5 additional factor of safety. The strength factor (F_s) is hence revised as 1.8×1.5 = 2.7.

The RRS is calculated as the floor response spectra at the level of equipment mounting using the ASCE 4-16 procedure for the DBE shaking.

F_R = exp(0.842 β_R) where β_R = 0.35

F_R = 1.34; this could be considered the “structural response factor”

Equipment response factor is obtained by accounting for the additional safety factors as follows:

Qualification method factor, F_{QM} = 1.0, β = 0.10

Damping factor, F_D = 1.29, β =0.08, considering design damping is 3% and median damping is 5%

Modeling factor, F_M = 1.0, β = 0.10

Mode combination factor, F_{MC} = 1.0, β = 0.10

Earthquake component combination factor, F_{ECC} = 1.0, β = 0.10

Total equipment response factor = 1.29

Total factor of safety = 2.7×1.34×1.29 = 4.68

Variability in equipment response = 0.21

Median ground acceleration capacity = 4.68×0.50 = 2.34 g

Total variability in ground acceleration capacity = $(0.13^2+0.35^2+0.21^2)^{1/2}$ = 0.43

HCLPF capacity = 0.86 g

4.3 Example 2—Shear Wall Design

The objective of this approach is to design a shear wall, as an example, using the design procedures in accordance with the ASCE 43 and ASCE 4 codes. The objectives are first to explore and understand the design and fragility differences that may result from using the different combinations of SDC and LS categories, and second to understand how these designs compare with the current approaches.

A simple shear wall was selected for this analysis for the following reasons:

1. Shear walls are major structural elements in nuclear power plant structures used to resist seismic loads;

2. A simplified shear wall represents a common design element that can be used to evaluate the various combinations of SDC and LS and resulting fragilities within the ASCE 43 and ASCE 4 design framework;

3. In several past and recent SPRAs, shear wall failures under seismic loads have been a significant contributor to both seismic core damage frequency (SCDF) and seismic large early release frequency (SLERF). Furthermore, failures of shear walls under seismic loads have been a singleton failure that directly leads to SCDF, and in a few cases to SLERF;

4. Results from this simplified problem can provide useful insights into how to adjust fragilities in existing SPRAs to incorporate the various combinations of SDC design ground motions and damage limit states.

4.3.1 Shear Wall Characteristics and Analysis Assumptions

1. The example shear wall is located on a hard rock site (Site A in Chapter 3), and thus there is no need to account for soil-structure interactions.

2. The rock site mimics a hard rock site in the Central Eastern US, for a nuclear power plant that has an existing probabilistic seismic hazard analysis (PSHA) and various DRS (Figure 3-7) developed during the recent Near-Term Task Force (NTTF) Recommendation 2.1 SPRA updates;

3. The shear wall dimensions are typical for internal shear walls in a nuclear power plant, with an aspect ratio less than or equal to two;

4. The desirable initial shear wall resonant frequency is between 5.0 Hz and 8.0 Hz, preferably closer to 5.0 Hz. To get the desired fundamental frequency and substantial in-plane shear forces additional mass was placed at the top of the wall;

5. Only in-plane failure modes and designs were explored; the top mass was assumed to be restrained for the out-of-plane motion;

6. Only in-plane and vertical excitations were considered. The design motions are in accordance with the DRS in Figure 3-7, and

7. The height and width dimensions of the shear wall were fixed, but the reinforcement ratios and thicknesses were varied to account for various combinations of SDCs and LSs.

4.3.2 ASCE 43 SDC and Limit States Evaluated

The following ASCE 43 SDC and LS categories were evaluated:

1. SDC–5 and LS-D
2. SDC–5 and LS-C
3. SDC–4 and LS-D
4. SDC–4 and LS-C
5. SDC–3 and LS-D
6. SDC–3 and LS-C
7. RG 1.60 (NRC, 2014) spectrum anchored to site SSE PGA and traditional design criteria (LS-D)

The combinations 1 through 6 of SDCs and LSs, which are consistent with the discussion in Chapter 3, are more likely to be used in an actual application than other combinations. Case 7 could be used in future work to help evaluate the differences in the design and fragilities using design ground motions and design criteria that have been applied to the design of many operating reactors. The seventh case provides an insight into how to adjust the existing fragility values to account for alternate design basis ground motions.

The fragility calculations were performed considering in-plane shear failure using the separation of variable approach in accordance with the Electric Power Research Institute (EPRI) guidance report TR-103959 (EPRI, 1994).

4.3.3 Design and Calculation Procedures

The following list summarizes some key steps and pertinent sections of ASCE 4 and 43 codes that were followed in the analyses:

1. Design standards: ASCE 4 (Chapters 1 through 4), ASCE 43 (Chapters 2 through 5), and ACI 349-13 (ACI, 2013)
2. Rock site selection: Site A of Table 3-3, Chapter 3 of this report
3. Modeling considerations: 2D finite element model, fixed base (no SSI or incoherency), and response spectrum analysis (Chapter 3 of ASCE 4).
    - Desired fundamental frequency between 5 and 8 Hz.
    - Damping: corresponding to the response level for the chosen limit state (Table 3.1 of ASCE 4)
    - Stiffness Calculations: Table 3.2 of ASCE 4

— Modeling of mass: ASCE 4 Section 3.4

— Shear walls: ASCE 4 Section 3.8.3

4. Design Ground Motions: DRS Site A shown in Figure 3-7
   A. SSDRS-5 spectra;

   B. SDDRS-4 spectra, and

   C. SSDRD-3 spectra.

   Two ground motion components were used: in-plane horizontal and vertical. The appropriate ratio for vertical to horizontal ground motion that was used for the specific plant and site was also used for this case study.

5. Analysis method: Chapter 4 of ASCE 4
   — Response spectrum analysis: Section 4.3 of ASCE 4

6. Design according to the following Sections of ASCE 43 and provisions for earthquake design in ACI 349-13
   — Sections on Evaluation of Seismic demand

   — Sections on Structural Capacity

   — Sections on Load Combinations and Acceptance Criteria

7. Fragility methodology: separation of variables (SOV)

   The fragility calculations were performed using the SOV approach considering three failure mechanisms (diagonal shear cracking, flexure, and shear friction) and a functional drift criterion of 0.007 in accordance with EPRI guidance documents (EPRI, 1994).

8. The probability of failure was computed by convolving hazard curves and fragility curves for various SDC and LS combinations.

4.3.4 Results, Insights, and Potential Future Activities

The following dimensions were chosen for the wall element:

Height (H) = 15 feet

Length (L) = 30 feet

Base case thickness (Th) = 2 feet

The design associated with the SDC–5 and LS-D is considered the base case and the other cases were evaluated in the context of this case. As stated earlier, the only variables that change with the different SDC and LS combinations are thickness and percentage of steel. The functional requirement for the wall to support a vertical load is unchanged. Tables 4-1 and 4-2 show results of design and fragility computations for the six SDC and LS cases in Section 4.3.2 along with details of basic parameters of shear walls and input PGA levels. Table 4-1 shows the results for 2-ft wall thickness. The parameters in the Table 4-1 are described as follows:

$\rho_H = \rho_V$, steel rebar to concrete cross sectional ratio in horizontal and vertical directions[3]; A_m, median capacity ($C_{50\%}$); βc, composite variability; HCLPF, high confidence low probability of failure ($C_{1\%}$); P_F, probability of failure evaluated by convolving the fragility curves with the mean hazard curve (PGA) for Site A in Figure 3-2. The performance of the shear walls is compared to the base case (the base case is 2-ft thick wall designed using the SDC–5 and LS-D combination). The Table 4-2 shows results of the same calculation but for a shear-wall thickness of 1.5 feet for cases for combinations of SDC and LS. The results are compared with the base case results in Table 4-1.

Table 4-1 Fragility and performance of shear wall (L=30 ft, H=15 ft, Th=2.0 ft) for combinations of SDC and LS using inelastic absorption factor Fμ calculated using EPRI (1994) methodology[4]

	SDC–5 LS-D (Base Case)	SDC–5 LS-C	SDC–4 LS-D	SDC–4 LS-C	SDC–3 LS-D	SDC–3 LS-C
PGA (g)	0.5	0.5	0.25	0.25	0.15	0.15
$\rho_H = \rho_V$	0.0141	0.0073	0.0055	0.0036	0.0033	0.0025
A_m (g)	3.07	2.92	2.83	2.71	2.7	2.64
βc	0.43	0.45	0.46	0.47	0.48	0.47
HCLPF (g)	1.13	1.05	1.00	0.93	0.91	0.90
P_F	1.11×10^{-6}	1.34×10^{-6}	1.51×10^{-6}	1.77×10^{-6}	1.83×10^{-6}	1.93×10^{-6}
Ratio P_F/P_F (base case)	1	1.2	1.36	1.60	1.65	1.74

Table 4-2 Fragility and performance of shear wall (L=30 ft, H=15 ft, Th=1.5 ft) for combinations of SDC and LS using inelastic absorption factor Fμ calculated using EPRI (1994) methodology

	Thickness = 2ft (Base Case)	Thickness = 1.5 ft				
	SDC–5 LS-D	SDC–5 LS-C	SDC–4 LS-D	SDC–4 LS-C	SDC–3 LS-D	SDC–3 LS-C
PGA (g)	0.5	0.5	0.25	0.25	0.15	0.15
$\rho_H = \rho_V$	0.0141	0.0090	0.0067	0.0045	0.0040	0.0027
A_m (g)	3.07	2.54	2.45	2.34	2.32	2.25
βc	0.43	0.44	0.45	0.43	0.46	0.47
HCLPF (g)	1.13	0.92	0.87	0.86	0.80	0.77
P_F	1.11×10^{-6}	1.94×10^{-6}	2.19×10^{-6}	2.37×10^{-6}	2.64×10^{-6}	2.9×10^{-6}
Ratio P_F/P_F(base case)	1.00	1.74	1.97	2.13	2.37	2.61

[3] A more precise definition of ρ_V is the ratio of the cross section of the vertical rebars (vertical shear reinforcement area) to the gross area of the horizontal section of concrete. A symmetric definition applies to ρ_H, for horizontal rebars and their total cross section compared to the gross area of the vertical section of concrete.

[4] The Fμ (inelastic absorption factor) value used results from applying EPRI guidance. In Section 4.2 a generic value of Fμ was used. The generic value is significantly different than the specific values calculated for the example shear walls. An additional sensitivity study was conducted to assess these differences and documented in a scientific notebook 1331e.

At this point no generic conclusions should be drawn from the example of a simple structural element. Section 4.4 considers various cases of structural and equipment fragilities in a broader sense. However, certain insights can be gained:

(i) While reductions in required steel and thicknesses are possible, these reductions may be limited by the other loads that a wall must withstand and other functional considerations;

(ii) There are reductions in the median values compared to the base case; however, as expected, because of the large uncertainties associated with the seismic hazard, probabilities of failure are not that sensitive; and

(iii) For this simple shear wall problem, the benefits of choosing a lower SDC or a lower limit state are comparable when compared to the base case. However, in actual design situations keeping the limit state as LS-D and selecting an alternate SDC is much easier to implement.

It is also instructive to compare the shear wall fragilities resulting from the generic approach of Section 4.2 with the specific design cases of this section. Table 4-3 shows this comparison.

Table 4-3 Comparison of generic fragility and design-specific fragility parameters

	SDC–5 and LS-D		SDC–5 and LS-C		SDC–4 and LS-D	
	Generic Fragility	Design-specific Fragility	Generic Fragility	Design-specific Fragility	Generic Fragility	Design-specific Fragility
A_m (g)	2.9 g	3.07g	1.93 g	2.92 g	1.45 g	2.83 g
β_c	0.43	0.43	0.43	0.45	0.43	0.46
HCLPF (g)	1.08 g	1.13 g	0.72 g	1.05 g	0.53 g	1.00 g

The generic fragility calculations in Section 4.2 assume that the overall seismic design, and hence the median capacity, is governed by the seismic loads. This is evident in the values obtained for SDC–4 and LS-D comparisons. The generic median capacity for SDC–4 is exactly half of the SDC–5 median capacity because the design PGA is exactly half. On the other hand (for this case where thickness is not changed), the design-specific fragility parameters are much closer for both SDC–5 and SDC–4. Therefore, it is important to note that the use of generic fragilities derived from ASCE 4 and ASCE 43 assumptions (Section 4.2) may show greater benefit, but also may overestimate the risk compared to the actual plant design fragility values (these observations do not necessarily apply to the generic fragilities that are based on past experience).

Based on the outcome from the example of a simple shear wall, additional activities are proposed as future activities:

(i) Compute limit states performance targets frequencies for the cases analyzed by adjusting median capacities and using beta values from the fragility analysis to provide insights into adjusting the SPRA fragilities as discussed in Section 4.4;

(ii) Analyze additional shear wall examples with alternate aspect ratios, to generate broader insights;

(iii) Investigate a simple three-dimensional structural system to assess the impact on floor response spectra for various SDC and LS combinations, and the feasibility of using limit states other than D in a more complex situation; and

(iv) Perform non-linear analyses of the shear wall using alternative approaches identified in ASCE 4 and ASCE 43 to evaluate the actual ductility demands and resulting drifts when LS-C is used.

4.4 Example 3—Progressive Use of SPRAs

To explore the implementation of the LMP/ASCE 43 Integration Approach described in Chapter 3 and to demonstrate the overall feasibility of using alternate SDC and limit state categories from ASCE 43, available SPRAs can be examined. The SPRAs can be used to evaluate the risk impact of using the alternate design categories and to examine the dominant sequences and develop criteria for selecting potential licensing basis event sequences (or events), as proposed in EPRI 18-04 (EPRI, 2018).

Ideally, both the currently available LWR [or advanced light-water reactors (ALWR)] SPRAs and SPRAs for an advanced reactor design should be used to fully explore implementation issues. Applications of the seven-step process to LWR or ALWR SPRAs (Section 4.4.1) and advanced reactors SPRAs (Section 4.4.2) are examined to identify potential detailed future analyses that would support the development of technical basis in support to an enhanced NRC regulatory guide for seismic safety.

Although the LMP framework has been developed for non-light-water reactor designs, a detailed exploratory example considering the LWR SPRAs is important to provide robust implementation insights. A second exploratory example that uses an advanced reactor SPRA would allow the identification of issues such as how to implement the F-C Target criteria and how to identify LBEs. The end states of a SPRA are different for LWR SPRAs and advanced reactor SPRAs, and unique considerations that are unique to the advanced reactors may be identified by exercising SPRAs.

In Example 3 the development of methods to recompute or adjust fragilities of SSCs for alternative combinations of SDC/LS pairs is explored, starting from fragilities associated with ASCE 43 SDC–5/LS-D. If such adjustments are feasible, existing SPRAs can be used to evaluate the risk impact of different SDC and LS combinations more fully. Potential approaches to adjust fragilities are discussed in this Example 3.

The details of available information constrains the nature and robustness of insights that can be gained. Because these are still in early design stages, the available information for advanced reactor designs will be for the design certification stage and therefore will contain less detail compared to an existing LWR SPRA. However, even this limited information will allow the NRC, NRC contractors, and LMP technical engineers to examine potentially unique aspects of SPRAs associated with the design stage within the risk-informed and performance-based (RIPB) framework.

Exploratory examples in this section are drawn from information and experience described in NUREG/CR-7214 (Budnitz and Mieler, 2016).

4.4.1 Use of LWR SPRAs

The selection of existing recent LWR SPRAs that have been used to support a 50.69 application would facilitate the exploration of the implementation of the seven-step for the following reasons:

1. The current SPRAs use the most recent PSHA and UHRS spectral shapes. These same PSHA and UHRS shapes are shown in Chapter 3. The use of UHRS as spectral shapes alleviates the issue of adjustment of fragilities for different design basis ground motions;

2. Extensive and detailed seismic response and fragility analyses have been carried out for several SPRAs. Detailed fragilities would facilitate the process of adjusting them for a different design;

3. The fragilities are more realistic and plant-specific, thus more amenable to use when considering questions of the impact of alternate design approaches;

4. These SPRAs include more accurate human error probabilities (HEPs) and non-seismic failures; and

5. For the plants that have implemented the 50.69 process, the classification of SSCs in various RISC categories is already available.

Before discussing how the seven-step process would be applied to LWRs, it is instructive to understand an approach used in NUREG/CR-7214 and results from a case study in that document. Two case studies were carried out in NUREG/CR-7214. Only one of the case studies (Plant A in NUREG/CR-7214) is summarized, and the reader is referred to the NUREG/CR-7214 for complete details. The case study analyzed an operating NPP located in the eastern United States that had available a relatively recent SPRA. The SPRA represents the baseline configuration of the plant as it was at the time of the SPRA analysis. Because the baseline configuration is based on the design of an operating NPP, it is assumed to satisfy all the requirements of the current NRC nuclear regulatory basis. In other words, the analysis and design processes are very similar to that of ASCE 43 and for SDC–5 and LS-D. One difference is the design ground motion. The NUREG/CR-7214 also studied two other configurations based on adaptations or reconfigurations of the baseline configuration. In particular, the second configuration adjusted the baseline design so that all SSCs satisfy the provisions of ASCE 43-05 SDC–5 and LS-D (i.e., that each SSC designed according to the standard has at least an annual probability of failure from earthquakes less than 1×10^{-5}). This is a conservative assumption because in the ASCE 43 approach the performance target for a LS associated with the SDC–5 itself is 1×10^{-5} /yr. The actual functional failure[5] probability will be less. The third configuration dealing with the defense in depth considerations is not discussed here.

Table 4-4 lists and describes the SSCs included in the analysis of Plant A. SSCs are categorized into three general groups: structural, mechanical, and electrical. Table 4-4 lists basic fragility parameters for each SSC, including median seismic capacity, Am, and uncertainty parameters, βr and βu. Table 4-5 lists and describes the dominant seismic event sequences

[5] The functional failure leads to a failure of required safety function as opposed to failure of the performance target (corresponding to exceeding a design limit state). Because of the conservatisms associated with the design process, the functional failure probabilities are much smaller than the probability of exceeding the limit states (performance target probabilities). The functional failure fragilities are used in a SPRA.

included in the analysis of Plant A as derived from the SPRA. These sequences are combinations of failures of the SSCs listed in Table 4-4. Together, Table 4-4 and Table 4-5 describe the baseline configuration of Plant A. This baseline configuration represents the NPP as it was originally designed and analyzed, therefore it is assumed to satisfy the provisions of the regulatory basis that were in effect at that time. In total, there are 8 SSCs and 9 seismic event sequences in the analysis of Plant A. Table 4-4 and Table 4-5 also summarize the results of a simplified SPRA of the baseline configuration. The seismic core damage frequency (CDF) for the baseline configuration of Plant A is 3.04×10^{-5}/yr. This number is referred to as the baseline seismic CDF for Plant A. Note that a single seismic event sequence, SEQ1, contributes disproportionately to the plant’s seismic risk profile, accounting for approximately 40% of the seismic CDF.

Table 4-4 Properties of SSCs in the baseline configuration of Plant A

SSC	Am (g)	βr	βu	HCLPF (g)	Pf
MECH1	0.77	0.25	0.22	0.35	2.54×10^{-6}
MECH2	0.68	0.18	0.32	0.30	4.18×10^{-6}
STRUC1	0.32	0.07	0.20	0.20	2.62×10^{-5}
MECH3	0.13	0.24	0.32	0.05	1.75×10^{-4}
MECH4	0.68	0.30	0.30	0.25	5.04×10^{-6}
ELEC1	0.30	0.27	0.40	0.10	4.62×10^{-5}
ELEC2	0.69	0.23	0.36	0.26	4.88×10^{-6}
STRUC2	0.53	0.23	0.42	0.18	1.20×10^{-5}

Table 4-5 Properties of dominant seismic sequences in the baseline configurations of Plant A

Sequence	HCLPF (g)	Pf	% of Seismic CDF
SEQ1	0.17	1.20×10^{-5}	39.37%
SEQ2	0.34	3.14×10^{-6}	10.32%
SEQ3	0.36	2.85×10^{-6}	9.35%
SEQ4	0.38	2.79×10^{-6}	9.18%
SEQ5	0.34	2.40×10^{-6}	7.88%
SEQ6	0.38	2.00×10^{-6}	6.59%
SEQ7	0.40	1.97×10^{-6}	6.48%
SEQ8	0.42	1.74×10^{-6}	5.73%
SEQ9	0.45	1.55×10^{-6}	5.10%
CDF	—	3.04×10^{-5}	100.00%

Figure 4-1 reproduced from the NUREG/CR-7214 plots the sensitivity of the seismic CDF to changes in the seismic capacities of individual SSCs. More specifically, it plots the absolute change in seismic CDF produced by changing the median capacity of each individual SSC, one at a time. Some SSCs have more impact on the plant’s seismic CDF than others. The impact of a particular SSC is a function of several factors, including its overall strength/capacity (as reflected by its fragility parameters) and also its particular role in the overall system (i.e., in the event sequence or sequences in which it participates).There are several observations that emerge from Figure 4-1. First, strengthening the most fragile SSC in the analysis, MECH3, has almost no impact on the plant’s seismic CDF, implying that simply strengthening the weakest SSCs in the plant may or may not be the most effective way to improve overall plant safety. In the context of the focus of this report on use of ASCE 43 approaches, this observation also implies a possibility of designing to a lesser SDC and LS compared to the current design. Second, the plant’s seismic CDF is disproportionately sensitive to changes in the strength of

STRUC2. A 30% decrease in median capacity produces a 60% increase in seismic CDF, meaning that an error in the design, analysis, operation, or maintenance of STRUC2 that reduces its capacity can significantly impact the safety of the plant. The second observation also provides insights into the question of using SDC categories and LS other than SDC–5 and LS-D. The portion of the curves which reflects reductions in the median capacities from the baseline configuration is of particular interest as use of alternate SDC and limit states will result in a reduction in the median capacities. Larger than 30% change in median capacities can be expected for alternate SDC and LS when compared to SDC–5 and LS-D median capacities shown in Figure 3-7 through Figure 3-9, and examples described in Section 4.2 and 4.3.

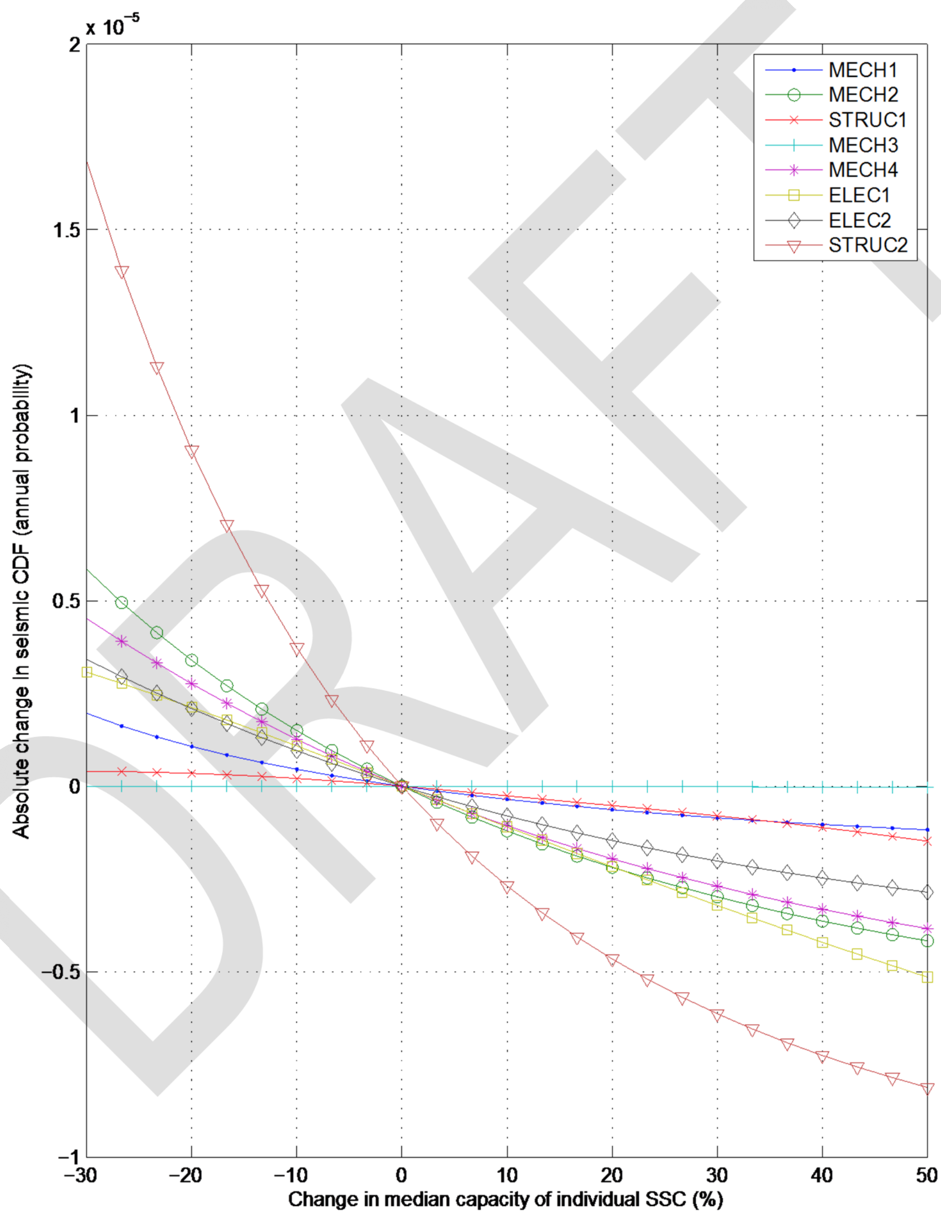

Figure 4-1 Sensitivity of the seismic CDF to one-at-a-time changes in the median seismic capacities of individual SSCs in the baseline configuration of Plant A. The ordinate is a frequency in units of 10^{-5} per year.

NUREG/CR-7214 also examined an alternate configuration by adjusting the fragilities of a small number of SSCs that had an annual probability of failure greater than 1×10^{-5}. There are four such components in Table 4-4. In this approach, the median seismic capacities of these components were increased until their annual failure probabilities are less than 1×10^{-5}. In the NUREG/CR-7214, this is denoted as the “ASCE 43-05 configuration” to reflect the fact that the performance target for the SDC–5 calls for the annual frequency of exceeding a limit state to be less than 1×10^{-5}. This is not an actual functional failure probability. However, in some sense this puts a lower bound on median capacity. Table 4-6 shows the revised fragilities of the four components.

Table 4-6 Changes in seismic capacities of SSCs in the “ASCE 43-05” configuration of Plant A

SSC	Baseline Am (g)	Updated Am (g)	% change In Am	Baseline Pf	Updated Pf	% change In Pf
STRUC1	0.32	0.45	40.6%	2.62E-5	9.91×10^{-6}	−62.2%
MECH3	0.13	0.53	307.7%	1.75E-4	9.58×10^{-6}	−94.5%
ELEC1	0.30	0.57	90.0%	4.62E-5	9.99×10^{-6}	−78.4%
STRUC2	0.53	0.57	7.5%	1.20E-5	9.88×10^{-6}	−17.5%

Overall, as a result of these changes, the plant’s seismic CDF decreases 35% from 3.04×10^{-5} to 1.98×10^{-5}.

In using the NUREG/CR-7214 approach for the purposes of integrating the seismic design with the RIPB framework and demonstrating the feasibility and validity of the stepwise process discussed in Chapter 3, the following additional factors should be considered:

(1) The baseline fragilities need to be adjusted so that the revised fragilities reflect a design carried out using an alternate SDC and LS.

(2) The fragilities of all the affected components need to be included in the SPRA simultaneously rather than examining only one-at-a-time changes in the fragilities of individual SSCs;

(3) The SPRA model should have non-seismic failures and HEP that are plant-specific; and

(4) The entire SPRA model needs to be analyzed to assess the identification of additional sequences, changes in the dominant sequences, and changes in the dominant contributors. This is necessary to identify whether the assigned SDCs and LSs are acceptable or need to be changed.

The following discussion of how the seven-step approach is implemented to demonstrate feasibility and validity assumes the availability of recent SPRAs (ideally, a PWR and a BWR SPRA) that have also been used for a 50.69 application. This is the most desirable situation; however, it may be possible to proceed with some lesser information, albeit such may not yield all the relevant insights.

Step 1—Initial Selection of the SDC and LS.

Because of the availability of several 50.69 analyses, the original selection of SDCs and LS can be done using the RISC categories shown in Figure 3-6. For example, for SSCs belonging to RISC-1, SDC–5 and LS-D may be chosen. For SSCs belonging to RISC-2, SDC–5 and LS-C (or SDC–4 and LS-D) may be chosen. In any case, this initial selection is much better informed because of the RISC.

Step 2—Seismic Design

This step is not applicable for the proposed feasibility demonstration. The fragilities would be adjusted to reflect SDC and LS as discussed in Step 3.

Step 3—Fragility Determination

Availability of the detailed and more plant specific fragilities would facilitate the adjustments of the fragilities. Examples and approaches discussed in Sections 4.2 and 4.3 provide some guidance on adjustment of fragilities. One simple and approximate procedure conceptually shown in Figure 4-2 could be explored in this progressive approach to develop some generic adjustment rules using the recent SPRAs that used UHRS for the structural shape.

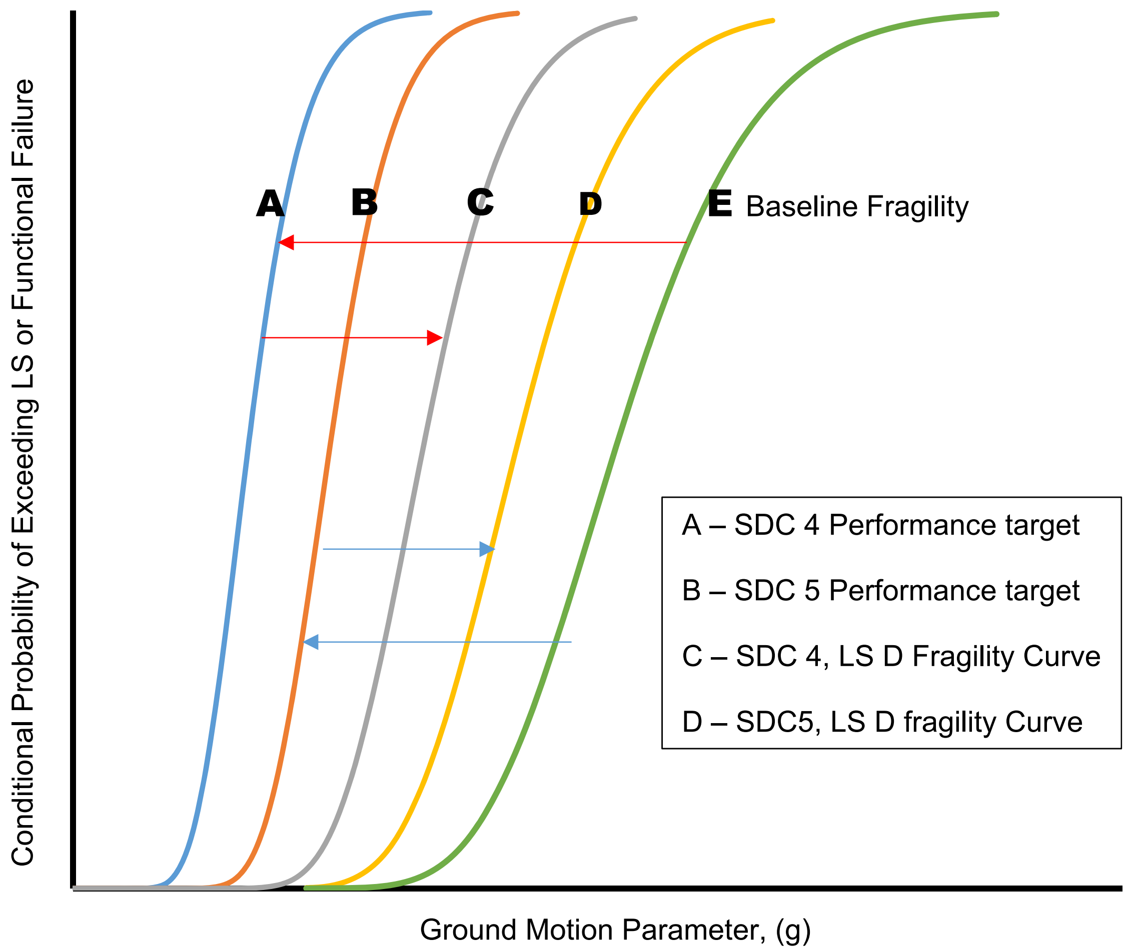

Figure 4-2 Illustration of fragility adjustment approach

For example, to develop a fragility curve that reflects SDC–5 and LS-D design of a particular component that was originally designed to different criteria, consider the following steps. In this procedure the baseline fragility (curve E in Figure 4-2) is available:

(1) Back calculate a curve associated with the probability of exceeding the limit state from the baseline fragility curve E (see footnote 5 that explains distinction between the two curves). Obtain curve B from curve E, where curve B represents a curve that is a conditional probability of exceeding the limit state for a given ground motion level for a particular component and baseline curve E, as stated earlier is the fragility curve of the same component from the SPRA and represents functional failure. The curve B is a performance target curve reflecting SDC–5. This is accomplished by calculating A_{mLS} (i.e., ground motion value associated with the median probability of exceeding LS) from the baseline A_m by removing non-applicable median safety factors and associated variabilities. That is, for LS-D, only retain terms associated with elastic response and strength;

(2) Convolve the site hazard curve and limit state fragility curve B to see if the performance target is met. If not, adjust A_{mLS} until the desired performance is achieved. For example, for the SDC–5 this target is an annual probability of 1×10^{-5} consistent with the ASCE 43 assumption as discussed in Section 4.2 or some other value considering component specific situation as discussed in Section 4.3. In real designs that will be based on ASCE 43 and ASCE 4, in general, the actual performance target values will be less than 1×10^{-5} and, in theory, the value of 1×10^{-5} is an upper bound.

(3) Using the revised A_{mLS}, recalculate the functional fragility to be used in the SPRA by considering the applicable safety factors in the original baseline analysis (curve E). For the example of SDC–5 and LS-D, the result is curve D in Figure 4-2. Although, in the figure, the curve D is shown on the left of the baseline curve, it could also be on the right side in some cases. The four components shown in Table 4-6 represent such a case.

(4) Use as similar approach to obtain a fragility curve for SDC–4 and LS-D for the same component as shown in Figure 4-2, curves A and C. Note that curve A is to the left of curve B as the performance target probability for SDC–4 is 1×10^{-4}, compared to 1×10^{-5} for SDC–5. Correspondingly, the resulting fragility curve, curve C is left of curve D.

The above procedure may be benchmarked against the analyses in Sections 4.2 and 4.3. For demonstration purposes and for understanding relative impacts of choosing different design categories and limit states, it is not necessary to use the most accurate fragilities for other SDC and LS categories. It is possible to estimate the approximate potential changes in the median capacities and still obtain robust insights in terms of what is feasible and what is not.

Step 4—Perform Seismic PRA Using a Progressive Approach

Although, in the actual application, the entire SPRA model should be rerun with inclusion of the revised fragilities, for the demonstration a progressive approach may be used to obtain better insights and make necessary adjustments as progress is made.

In this progressive approach, instead of exercising the entire SPRA model, the examination starts with the most contributing top ten or so seismic event sequences as done for Plant A in NUREG/CR-7214. In many LWR SPRAs there are single failures, such as a building failure, which lead to core damage sequences. In such a case, it may be relatively straightforward to modify the fragility of that single component as if it were designed to the different SDC and LS.

The next steps are to look progressively at complex sequences that include components from different RISC categories and non-seismic failures and HEPs. The fragilities of components in different risk categories would be revised using the approach described in Step 3 and recomputing the event sequence frequencies.

Implementing this exercise would provide insights into the proposed seven-step process and may reveal some inherent constraints and limitations that would apply for a practical application.

Eventually, the full SPRA model should be run to examine how the wholesale changes of fragilities influence event sequences, dominant contributors, and other insights, such as those related to defense in depth considerations. As the LBE and safety classification are derived from PRAs in the RIPB framework, this is a very important step.

Step 5—Check the proposed classification against the risk, defense-in-depth, and other qualitative criteria

For the LWRs, the end states of CDF and LRF can be adopted as applicable risk criteria to examine whether the design categories need to be revised and whether the RISC classification needs to be revised for some SSCs.

Step 6—Iteration, and Step 7, Final Classification would be implemented as described in Chapter 3.

4.4.2 Use of Advance Reactor PRAs

There are several designs of advanced reactors underway, and it is understood that SPRAs associated with some of the designs exist (for example, for the standard modular high temperature gas cooled reactor). Among several differences between an advanced-reactor SPRA and an LWR SPRA, it is expected that an advanced reactor design will have fewer safety/risk significant SSCs because of the passive nature of the design among other reasons. On the other hand, the end states of a PRA for some of the advanced reactors are dose consequences and cumulative risk rather than the CDF and large early release fraction (LERF) for LWRs. Subject to the availability of an advanced reactor SPRA, the following discussion highlights considerations and approaches involved in demonstrating feasibility and validity of the seven-step approach. The availability of software associated with the quantification of event sequences will also be necessary as the end states are dose consequences. In general, consultation with the advanced-reactor PRA team in the beginning will facilitate this undertaking.

Step 1—Initial Selection of the ASCE 43 SDC and LS Categories

Compared to the LWR case, this step is a little more complex in the sense that the fragilities used in the advanced reactor SPRAs may have some generic fragilities or may have lesser details. Some judgment may be required to assign different SDC and LS to selected SSCs. However, because of the fewer SSCs and because a high degree of precision is not needed for demonstration purposes, several variations can be examined.

Step 2—Seismic Design

This step is not applicable for the feasibility and validity demonstration. The fragilities may be adjusted to reflect SDC and LS as discussed in Step 3.

Step 3—Fragility Determination

The approach described in Section 4.4 may be used. There may be need for some additional judgments because of the lack of details and potential use of generic fragilities.

Step 4—Perform Seismic PRA

This is a more complex step compared to LWR SPRAs because of the different end states and limited experience with these types of SPRAs. The progressive approach described in connection with the LWR may also be implemented. Eventually, the full SPRA model should be exercised to evaluate how the wholesale changes of fragilities influence event sequences, dominant contributors, and other insights, such as those related to defense in depth considerations.

Step 5—Check the proposed classification against the risk criteria

Results would be compared to the applicable risk criteria of the F-C Target to identify whether the design categories need to be revised for some SSCs.

Steps 6 and 7 are anticipated to be the same as for an LWR.

4.4.3 Summary

Full demonstration examples are expected to provide not only insights on the feasibility and validity of using ASCE 43 and ASCE 4 design approaches, but also potential benefits in terms of a more balanced design, more uniform safety margins, more effective and better understood defense-in-depth, and cost savings. A complete analysis should provide insights into related regulatory and practical considerations and potential revisions of the current guidance.

5 SUMMARY, CONCLUSIONS, AND NEXT STEPS

The work described in this report examines concepts and processes to align the licensing modernization project (LMP) initiative within the existing risk-informed and performance-based (RIPB) framework for seismic safety and to identify future potential activities. This report also contributes to the technical bases that could support potential publication of a regulatory guide on the proposed approach.

The following activities were conducted in this project to support the project objectives:

- Selected regulations and guidance that are pertinent to implementation of the ASCE 43 seismic design approach were reviewed in order to identify any potential changes in current U.S. Nuclear Regulatory Commission (NRC) regulations or guidance that may be needed.

- The LMP framework, presented in the document U.S. Nuclear Energy Institute (NEI) 18-04 (NEI, 2018), was reviewed to evaluate how to align the American Society of Civil Engineers (ASCE) 43 seismic design approach within this initiative and within the larger RIPB framework. This approach is referred to as LMP/ASCE 43 Integration Approach. Two crucial factors in this alignment are to clearly define (i) how probabilistic risk assessments (PRAs) will be used in making design decisions and (ii) how the sequence-level and plant-level risk measures can be integrated in an iterative design process.

- ASCE Standards 1, 4, and 43 were reviewed to identify (i) the differences in the design process when compared to current methods, (ii) the interplay among the different seismic design categories and limit states (LSs) in meeting risk metrics [as opposed to the current approach of using a single seismic design category (SDC)–5, and LS-D], and (iii) how the explicit implementation of performance targets for individual structures, systems, and components (SSCs) could be implemented in a reasonable and practical seismic design process.

- A seven-step design process was formulated that incorporates the LMP concepts with the ASCE 43 seismic design approach in order to achieve the project objectives.

- Several implementation issues were assessed yielding initial insights about how a design process could be implemented. Potential impacts of the integrated seismic design process and the LMP framework on the broader regulatory and operational requirements of nuclear plant safety were also identified.

- Site-specific ground motions from nine power plant sites in the Central and Eastern United States (CEUS), representing various site conditions (i.e., hard rock, stiff soil, and soft soil sites) and geographical areas, were selected to identify the benefits of the ASCE 43 seismic design approach. Ground motions corresponding to the top three SDCs in ASCE 43 were compared to the design ground motions derived from current seismic design requirements to demonstrate the benefits of the enhanced RIPB framework.

- A series of simple calculations were performed to support the development of the LMP/ASCE 43 Integration Approach to demonstrate its feasibility and validity, and to better identify implementation issues and gauge the level of effort needed to conduct potential future activities to support the overall LMP/seismic-safety integration goal. These calculations included (i) generic fragilities based on underlying assumptions used in ASCE 43 for alternate SDC and LS selections, (ii) designs of simple shear wall elements using alternate SDC and LS combinations for the same site to understand the impacts on physical designs and fragilities for these combinations, and (iii) development of a more detailed PRA-based demonstration analysis with progressive scope for future activities that includes exploring approximated adjustments to SSC fragilities to emulate different SDC/LS combinations when a detailed seismic probabilistic risk assessment (SPRA) and supporting information is available.

The report also describes and evaluates an initial set of practical steps that are needed to implement the LMP framework and related RIPB enhancements to existing seismic design process. These practical steps were then evaluated using the stylized seven-step seismic design process.

To demonstrate the potential benefits of relaxing the requirement that all safety-significant SSC need not be designed to SDC–5 ground motions, the ratios of peak ground acceleration (PGA) and 5 hertz (Hz) ground motions for SDC–4 and SDC–3 were compared to the SDC–5 ground motions for nine sites in the CEUS. These ground motions were derived from the licensee probabilistic seismic hazard analysis (PSHA) results submittals to the NRC in response to the Commission's request based on a recommendation developed by the NRC's Near-Term Task Force (NTTF) in the aftermath of the accident at Fukushima. (This recommendation required all licensees and certain other permit holders to compute a new ground motion using the present practices and guidance and the most recent earthquake data for each site.) The procedure used by the licensee to develop the PSHA and associated response spectra were based on Regulatory Guide (RG) 1.208 (NRC, 2007a) and ASCE 43 (ASCE/SEI, 2005). ASCE 43 was also used in this report to obtain the SDC–3 and SDC–4 PGA and 5 Hz ground motions for these comparisons. In summary, for these nine sites, SDC–4/SDC–5 average ratios for peak ground acceleration and 5 Hz spectral accelerations are close to 0.55, and SDC–3/SDC–5 average ratios are close to 0.35. These results show that substantial reductions in design ground motions can be achieved through selection of alternate SDCs. The report includes comparisons of entire spectra and derives more detailed site-specific insights.

The following are the main conclusions from the analyses and results described in this report:

- An initial seven step process has been developed for the LMP/ASCE 43 Integration Approach. This is a general process that can be used for both Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 and Part 52 applications. The process is also technology-inclusive.

- There are no obvious impediments in implementing the enhanced risk-informed seismic design requirements based on the LMP framework and recent updates to the several seismic design standards and regulatory guides. The report discusses several technical, programmatic, and regulatory considerations for the successful implementation of the seven-step process. These discussions also form a basis for more detailed feasibility demonstration activities that may be undertaken in the future.

- The major benefits of the proposed RIBP enhancements come from the flexibility to assign different SDCs (thus, different design basis ground motion levels) and different design performance limits (that is, different LSs) to SSCs considering their risk-significance and other risk-informed decision-making considerations (in contrast to the current approach that uses a single design basis earthquake and a very stringent elastic LS for all SSCs, irrespective of their risk significance). Thus, in this enhanced RIPB seismic design process, the safety margins of individual SSCs are controlled according to their contribution to system-level and plant-level risk, thereby reducing unnecessary conservatisms (or providing additional margins, as needed) and achieving a more risk-balanced design.

- The actual detailed design process is very similar to that being used currently, once the SDC (design basis ground motion levels) and LS are established for SSCs. The analyses and design processes that are being used are well practiced and rely on existing standards and guidance.

- The report contributes to a technical basis to develop a regulatory guide to implement the RIPB seismic design.

- Stakeholder interactions will be crucial for broader acceptance and refinement of the proposed process and for developing additional cooperative activities.

- Additional analyses are recommended to fully demonstrate the implementation and to identify associated pros and cons of the proposed changes to the seismic design process within the larger context of seismic design, seismic safety, optimized costs, and existing seismic safety regulations.

The report also includes several results that highlight the implications of some of the current regulatory requirements, such as those associated with the minimum earthquake design levels and earthquake shutdown and restart criteria. The detailed description of the seven-step process includes discussions of additional considerations and key management and technical considerations that are needed for efficient implementation of the enhanced RIPB concepts described in the report.

Future work is proposed to both demonstrate the feasibility and validity of the process developed so far through some simple SPRA models and to obtain and include stakeholder input through public workshops. Depending on the outcomes of the workshops, a cooperative effort with industry may be considered to conduct more detailed evaluations of the concepts and process through more detailed SPRA models. These more detailed evaluations will provide additional technical and regulatory insights into implementation issues and will further clarify the concepts and process described in this initial report.

The following activities should be considered for future work:

- Evaluate the seismic design of a small stylized system (going beyond a single shear wall) to more fully explore the adequacy of guidance in codes such as ASCE 43.

- Propose a cooperative effort with industry to conduct additional evaluations of the concepts and process through simplified and more detailed SPRA models. These activities will provide additional technical and regulatory insights into implementation issues and will further clarify the enhanced RIPB concepts described in this report. These cooperative activities could include:

- — An examination of a simplified and a detailed SPRA of an actual light water reactor (or advanced light water reactor) and implement the seven-step process to better assess advantages and limitations of the LMP/ASCE 43 Integration Approach to seismic safety.

- — An examination of an advanced reactor PRA (for example, a standard modular high temperature gas cooled reactor or a fast-spectrum sodium-cooled small modular reactor) and implement the seven-step process with the goal to identify changes in current seismic safety guidance.

- Communicate the results of the current work and discuss potential future work in a workshop or a series of workshops.

6 REFERENCES

60 FR 42662. Federal Register. “Rules and Regulations.” Vol. 60, No 158.

ACI. “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary.” ACI-349-13. Farmington Hill, Michigan: American Concrete Institute. 2013.

ANSI/AISC. “Specifications for Safety-Related Steel Structures for Nucellar Facilities.” ANSI/AISC N690-18. Chicago, Illinois: American National Standards Institute/American Institute of Steel Construction. 2018.

ANSI/ANS. “Probabilistic Seismic Hazard Analysis.” ANSI/ANS-2.29-2008. La Grange Park, Illinois: American National Standards Institute/American Nuclear Society. 2008.

ANSI/ANS. “Categorization of Nuclear Facility Structures, Systems, and Components for Seismic Design.” ANSI/ANS-2.26-2004 (R2017). La Grange Park, Illinois: American National Standards Institute/American Nuclear Society. 2004.

ASCE. “Seismic Analysis of Safety-Related Nuclear Structures and Commentary.” ASCE 4-98. New York, New York: American Society of Civil Engineers. 1998.

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

ASCE/SEI. “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities.” ASCE/SEI 43-05. New York, New York: American Society of Civil Engineers and Structural Engineering Institute. 2005.

Budnitz, R.J. and M.W. Mieler. NUREG/CR-7214, “Toward a More Risk-Informed and Performance-Based Framework for the Regulation of the Seismic Safety of Nuclear Power Plants.” ADAMS Accession No. ML16126A008. Washington, DC: U.S. Nuclear Regulatory Commission. May 2016.

Budnitz, R.J., D. Konstantinidis, Z. Zhou, and T.J. Herrity. NUREG/CR-7193, “Evaluations of NRC Seismic-Structural Regulations and Regulatory Guidance, and Simulation-Evaluation Tools for Applicability to Small Modular Reactors (SMRs).” ADAMS Accession No. ML15083A178. Washington, DC: U.S. Nuclear Regulatory Commission. March 2014.

Cornell, C.A. “Engineering Seismic Risk Analysis.” *Bulletin of the Seismological Society of America 58(5).* pp. 1,583–1,606. Erratum: 49(4). 1,733. 1968.

Cummings, G.E. NUREG/CR-4431, “Summary Report on Seismic Safety Margins Research Program.” Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission. 1986.

Dasgupta, B. “Evaluation of Methods Used to Calculate Seismic Fragility Curves.” San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2017. ADAMS Accession No. ML17122A268.

DOE. “Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities.” DOE–STD–1020–2016. Washington, DC: U.S. Department of Energy. 2016.

_____. “Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities.” DOE–STD–1020–94. Washington, DC: U.S. Department of Energy. 1994.

EPRI. “Methodology for Developing Seismic Fragilities.” EPRI TR-103959. Palo Alto, California: Electric Power Research Institute. 1994.

INL. “Licensing Modernization Project for Advanced Reactor Technologies: FY 2018 Project Status Report.” INL/EXT-18-46151. Idaho Falls, Idaho: Idaho National Laboratory. 2018

NEI. “Risk-informed Performance-based Guidance for Non-Light Water Reactor Licensing Basis Development.” Technical Report NEI 18-04. Washington, DC: Nuclear Energy Institute. 2018.

NRC. Draft Regulatory Guide 1353 (RG-1353), “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: U.S. Nuclear Regulatory Commission. 2019.

_____. NUREG–2213, “Updated implementation Guidelines for SSHAC Hazard Studies.” Washington, DC: U.S. Nuclear Regulatory Commission. 2018b (in press).

_____. Regulatory Guide 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.” Washington, DC: U.S. Nuclear Regulatory Commission. January 2018c. ADAMS Accession No. ML17317A256.

_____. NUREG–1520, “Standard Review Plan for Fuel Cycle Facilities License Applications.” Revision 2. Washington, DC: U.S. Nuclear Regulatory Commission. 2015.

_____. Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants.” Revision 2. Washington, DC: U.S. Nuclear Regulatory Commission. 2014.

_____. “Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident.” Enclosure to SECY–11–0093. Washington, DC: U.S. Nuclear Regulatory Commission. July 2011b. ADAMS Accession No. ML112510271.

_____. DC/COL-ISG-020, “Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors.” Washington, DC: U.S. Nuclear Regulatory Commission. March 15, 2010a. ADAMS Accession No. ML100491233.

_____. DC/COL-ISG-017, “Interim Staff Guidance on Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses.” Washington, DC: U.S. Nuclear Regulatory Commission. March 24, 2010b. ADAMS Accession No. ML100570203.

_____. DC/COL-ISG-01, “Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion.” Washington, DC: U.S. Nuclear Regulatory Commission. May 19, 2008. ADAMS Accession No. ML081400293.

_____. Regulatory Guide 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion.” Washington, DC: Nuclear Regulatory Commission. 2007a.

_____.SECY 07-0074, “Risk-Informed and Performance-Based Plan.” Washington, DC: Nuclear Regulatory Commission. April 26, 2007b.

_____. NUREG–1860, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing.” Washington, DC: U.S. Nuclear Regulatory Commission. December 2007c. ADAMS Accession No. ML080440170.

_____. “Risk-Informed Regulation Implementation Plan.” SECY 00-0062 and SECY 00-0213, Washington, DC: U.S. Nuclear Regulatory Commission. March 15, 2000 and October 26, 2000.

_____. NUREG/CR–6372, “Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts.” Washington, DC: U.S. Nuclear Regulatory Commission. 1997.

_____. “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement. 60 FR 42622. Washington, DC: U.S. Nuclear Regulatory Commission, August 16, 1995.

_____. SECY 94-219, “Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA).” Washington, DC: U.S. Nuclear Regulatory Commission. August 19, 1994.

_____. “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.” SECY 93-087. Washington, DC: U.S. Nuclear Regulatory Commission. April 2, 1993. ADAMS Accession No. ML003708021.

_____. NUREG–1407, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities.” Washington, DC: U.S. Nuclear Regulatory Commission. 1991.

_____. “Safety Goals for the Operation of Nuclear Power Plants: Policy Statement,” *Federal Register* 51, p. 30028. Washington, DC: U.S. Nuclear Regulatory Commission. 1986.

Risk Engineering Inc. NUREG/CR–6728, “Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines.” Washington, DC: U.S. Nuclear Regulatory Commission. 2001.

Stevenson, J.D., R.P. Kennedy, and W.J. Hall. “Nuclear Power Plant Seismic Design – A Review of Selected Topics.” *Nuclear Energy and Design*, Vol. 79, pp. 7–18, 1984.