JAPAN LESSONS-LEARNED PROJECT DIRECTORATE

JLD-ISG-2012-04

Guidance on Performing a Seismic Margin Assessment in Response to the March 2012 Request for Information Letter

Interim Staff Guidance
Revision 0

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INTERIM STAFF GUIDANCE
JAPAN LESSONS-LEARNED PROJECT DIRECTORATE

GUIDANCE ON PERFORMING A SEISMIC MARGIN ASSESSMENT IN RESPONSE TO THE MARCH 2012 REQUEST FOR INFORMATION LETTER

Purpose

The U.S. Nuclear Regulatory Commission (NRC) staff is providing this interim staff guidance (ISG) as supplemental guidance to nuclear power reactor licensees on an acceptable method for performing a seismic margin assessment (SMA) as referred to in the March 12, 2012, NRC letter entitled, "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendation 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," hereafter called the "50.54(f) letter."

This document describes the enhancements to the NRC SMA method originally described in NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," which are needed to meet the objectives of the 50.54(f) letter. This ISG presents staff positions on enhancements to the major elements of the NRC SMA and provides updated references for the use of recent advances in both methods and guidance, including guidance in the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS), "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Standard ASME/ANS RA-Sa-2009, and the screening, prioritization, and implementation document (SPID) currently under development by industry for NRC endorsement.

This guidance, at this time, is only intended to be used for an enhanced NRC-method SMA conducted in response to the 50.54(f) letter, and not for any other purposes. The NRC ISG DC/COL-ISG-020, "Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors," remains the NRC's current guidance for application of an SMA to new reactors licensing. A probabilistic risk assessment (PRA)-based SMA is acceptable as long as the licensee follows the guidance in this ISG, the SPID, and ISG DC/COL-ISG-020 (with the exception of elements that clearly and only relate to new reactor licensing). The contents of this ISG have no implications for NRC ISG DC/COL-ISG-020, the ASME/ANS probabilistic risk assessment (PRA) standard, or any other document.

Licensees may propose other methods for satisfying the 50.54(f) letter. The NRC staff will review such methods and determine their acceptability on a case-by-case basis.

Introduction

Following the events at the Fukushima Dai-ichi nuclear power plant in Japan on March 11, 2011, the NRC established a senior-level agency task force referred to as the Near-Term Task Force (NTTF). The agency tasked the NTTF with conducting a systematic and methodical review of NRC regulations and processes and determining if the agency should make additional improvements to these programs in light of the events at Fukushima Dai-ichi. As a result of this review, the NTTF developed a comprehensive set of recommendations, documented in SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated July 12, 2011. The NRC enhanced these recommendations through interactions with stakeholders. SECY-11-0124, "Recommended Actions To Be Taken without Delay from the Near-Term Task Force Report," dated September 9, 2011, and SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, document the staff's efforts.

In accordance with the staff requirements memorandum (SRM) for SECY-11-0093, the NRC staff reviewed the NTTF recommendations within the context of the NRC's existing regulatory framework and considered the various regulatory vehicles available to the agency to implement the recommendations. SECY-11-0124 and SECY-11-0137 established the staff's priorities for the NTTF recommendations.

In March 2012, the NRC issued a 50.54(f) letter. Enclosure 1 of that letter, "Recommendation 2.1: Seismic," described the actions related to seismic hazard and risk reassessments for licensees to take in response to the letter. The SMA method is among the approaches discussed in Enclosure 1, which may be appropriate for some plants depending on the outcome of the hazard reassessment phase.

Enclosure 1 to the 50.54(f) letter states that, "[t]he SMA approach should be the NRC SMA approach (e.g., NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," issued in August 1985 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090500182) as enhanced for full-scope plants in NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities"). The SMA approach should include both core damage (accident prevention) and large early release (accident mitigation)."

This document describes the enhancements to the NRC SMA method, originally described in NUREG/CR-4334, that are needed to meet the objectives of the 50.54(f) letter. In addition, this ISG presents staff positions on the major elements of the NRC SMA. This ISG also provides updated references for the use of more recent advances in methods and guidance, including guidance in the ASME/ANS standard and the SPID currently under development by industry for NRC endorsement.

Three methods currently can be used to perform an SMA:

(1) the PRA-based SMA method (as described in NRC ISG DC/COL-ISG-020)

(2) the NRC SMA method (as described in NUREG/CR-4334, supplemented by NUREG/CR-4482, "Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants," and NUREG/CR-5076, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants: The Importance of BWR [boiling-water reactor] Plant Systems and Functions to Seismic Margins")

(3) the Electric Power Research Institute (EPRI) SMA method (as described in EPRI NP-6041-SL Revision 1, “A Methodology for Assessment of Nuclear Power Plant Seismic Margin” (EPRI, 1991)).

These three methods differ in two key areas: the initiators considered in the analysis; and, the system logic model approach used.

This ISG addresses only the NRC SMA method. It does not address either the EPRI SMA method or the PRA-based SMA method. The EPRI SMA method is not acceptable for satisfying the objectives described in the 50.54(f) letter because of its use of success paths. In principle, the full PRA-based SMA can be used as long as the licensee follows the guidance in this ISG, the SPID, and ISG DC/COL-ISG-020 (with the exception of elements that clearly and only relate to new reactor licensing). In addition, ISG DC/COL-ISG-020 does not address some specific considerations pertinent to the 50.54(f) request (e.g. large early release frequency and mission time).

The NRC staff will use this ISG guidance when reviewing the technical adequacy of enhanced NRC-method SMAs submitted by licensees in accordance with the subject 50.54(f) letter. It shall remain in effect until it has been superseded or withdrawn.

Public meetings on the proposed SMA guidance document were held to gain stakeholder input before the guidance was formally issued for public comment. On September 10, 2012 (in the *Federal Register* (FR), 77 FR 55510), the NRC requested public comments on draft JLD-ISG-12-04. The comments received and staff responses are contained in “NRC Responses to Public Comments,” for JLD-ISG-2012-04, which can be found at ADAMS Accession No. ML12290A002.

Implementation

Except in those cases in which a licensee or construction permit holder proposes an acceptable alternative method for complying with the 50.54(f) letter, the NRC staff will use the methods described in this ISG to evaluate licensee submittals requested in the 50.54(f) letter.

Backfitting Discussion

Licensees may use the guidance in this document as one acceptable method for responding to a portion of the information requested in the 50.54(f) letter. Accordingly, the NRC staff’s issuance of this ISG is not considered backfitting, as defined in 10 CFR 50.109(a)(1).

Final Resolution

This ISG, or a portion thereof, may subsequently be incorporated into other guidance documents, as appropriate.

Attachment

“Guidance on Performing a Seismic Margin Assessment in Response to the March 2012 Request for Information Letter.”

References

American Society of Mechanical Engineers/American Nuclear Society, Standard ASME/ANS RA-Sa-2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” 2009.

Electric Power Research Institute, “A Methodology for Assessment of Nuclear Power Plant Seismic Margin,” EPRI Report NP-6041-SL, Revision 1, Palo Alto, California, 1991.

U.S. Nuclear Regulatory Commission, “An Approach to the Quantification of Seismic Margins in Nuclear Power Plants,” NUREG/CR-4334, August 1985 (ADAMS Accession No. ML090500182).

U.S. Nuclear Regulatory Commission, “Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants,” NUREG/CR-4482, 1986 (ADAMS Accession No. ML12069A017).

U.S. Nuclear Regulatory Commission, “An Approach to the Quantification of Seismic Margins in Nuclear Power Plants: The Importance of BWR Plant Systems and Functions to Seismic Margins,” NUREG/CR-5076, 1988.

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

U.S. Nuclear Regulatory Commission, “Recommendations for Enhancing Reactor Safety in the 21st Century, the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” Commission Paper SECY-11-0093, July 12, 2011 (ADAMS Accession No. ML11186A950).

U.S. Nuclear Regulatory Commission, “Staff Requirements – SECY-11-0093 – Near-Term Report and Recommendations for Agency Actions following the Events in Japan,” Commission Paper SRM-SECY-11-0093, August 19, 2011 (ADAMS Accession No. ML112310021).

U.S. Nuclear Regulatory Commission, “Recommended Actions to be Taken without Delay from the Near-Term Task Force Report,” Commission Paper SECY-11-0124, September 9, 2011 (ADAMS Accession No. ML11245A158).

U.S. Nuclear Regulatory Commission, “Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned,” Commission Paper SECY-11-0137, October 3, 2011 (ADAMS Accession No. ML11272A111).

U.S. Nuclear Regulatory Commission, “Staff Requirements – SECY-11-0124 – Recommended Actions to be Taken without Delay from the Near-Term Task Force Report,” Commission Paper SRM-SECY-11-0124, October 18, 2011 (ADAMS Accession No. ML112911571).

U.S. Nuclear Regulatory Commission, “Staff Requirements – SECY-11-0137 – Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned,” Commission Paper SRM-SECY-11-0137, December 15, 2011 (ADAMS Accession No. ML113490055).

U.S. Nuclear Regulatory Commission Letter to All Power Reactor Licensees et al., "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendation 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 12, 2012 (ADAMS Accession No. ML12053A340).

U.S. Nuclear Regulatory Commission, "Draft Interim Staff Guidance on Performing a Seismic Margin Assessment in Response to the March 2012 Request for Information Letter," JLD-ISG-12-04, September 4, 2012 (ADAMS Accession No. ML12222A327).

U.S. Nuclear Regulatory Commission, "NRC Responses to Public Comments: Japan Lessons Learned Project Directorate Interim Staff Guidance JLD-ISG-2012-04: Performing a Seismic Margin Assessment in Response to the March 2012 Request for Information Letter," dated November 16, 2012 (ADAMS Accession No. ML12290A002).

Enhancements to the U.S. Nuclear Regulatory Commission Method for Seismic Margin Assessment in Response to the March 2012 Request for Information Letter

1.0 Purpose

The U.S. Nuclear Regulatory Commission (NRC) staff is providing this interim staff guidance (ISG) as supplemental guidance to nuclear power reactor licensees on an acceptable method for performing a seismic margin assessment (SMA) as referred to in the March 12, 2012, NRC letter entitled, "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendation 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," hereafter called the "50.54(f) letter."

This document describes the enhancements to the NRC SMA method, originally described in NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," which are needed to meet the objectives of the 50.54(f) letter. This ISG presents staff positions on enhancements to the major elements of SMA and provides updated references for the use of recent advances in methods and guidance, including guidance in the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS), "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Standard ASME/ANS RA-Sa-2009, (hereafter called the "ASME/ANS PRA standard") and the screening, prioritization, and implementation document (SPID) currently under development by industry for NRC endorsement (see Section 4.1.2).

This guidance, at this time, is only intended to be used for an enhanced NRC method SMA conducted in response to the 50.54(f) letter, and not for any other purposes. The NRC ISG DC/COL-ISG-020, "Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors," remains the NRC's current guidance for application of an SMA to new reactors licensing. A probabilistic risk assessment (PRA)-based SMA is acceptable as long as the licensee follows the guidance in this ISG, the SPID, and ISG DC/COL-ISG-020 (with the exception of elements that clearly and only relate to new reactor licensing). The contents of this ISG have no implications for NRC ISG DC/COL-ISG-020, the ASME/ANS PRA standard, or any other document.

Licensees may propose other methods for satisfying the 50.54(f) letter. The NRC staff will review such methods and determine their acceptability on a case-by-case basis.

2.0 Key Terms and Concepts

This section defines key terms and concepts used in this ISG.

Accident Sequence–A representation in terms of an initiating event followed by a sequence of failures or successes of events (such as system, function, or operator performance) that can lead to undesired consequences, with a specified end state (e.g., core damage or early release).

Accident Sequence Analysis–The process to determine the combination of initiating events, safety functions, and system failures and successes that may lead to core damage or large early release.

Fragility–The conditional probability of the failure of a structure, system, or component (SSC) at a given hazard input level. For seismic fragility, the input parameter could be peak ground acceleration (PGA), peak spectral acceleration, floor spectral acceleration, or others. The fragility calculation typically uses a double lognormal model with three parameters, which are the median acceleration capacity (A_m), the logarithmic standard deviation of the aleatory (randomness) uncertainty in capacity (β_R), and the logarithmic standard deviation of the epistemic (modeling and data) uncertainty in the median capacity (β_U). The aleatory and epistemic uncertainty can be combined into a composite variability. The fragility using a composite variability is referred to as the mean fragility.

Ground Motion Response Spectra (GMRS)–The site-specific spectra characterized by horizontal and vertical response spectra determined as free-field motions on the ground surface or as free-field outcrop motions on the uppermost in situ competent material using performance-based procedures in accordance with NRC Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion."

High Confidence of Low Probability of Failure (HCLPF) Capacity–A measure of seismic ruggedness. HCLPF capacity is defined as the earthquake motion level at which there is a high (95 percent) confidence of a low (at most 5 percent) probability of failure of a single SSC or of an ensemble of them. It is formally defined (NUREG/CR-4334) using the lognormal fragility model as $A_m \exp[-1.65 (\beta_R + \beta_U)]$. When the logarithmic standard deviation of composite variability β_C is used, the HCLPF capacity can be approximated as the ground motion level at which the probability of failure is at most 1 percent. In this case, HCLPF capacity is expressed as $A_m \exp[-2.33 \beta_C]$. The conservative deterministic failure margin (CDFM) (methodology described in EPRI Report NP-6041-SL Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" (EPRI, 1991), produces a HCLPF capacity estimate directly, without developing the full fragility curve.

Large Early Release–The rapid, unmitigated release of airborne fission products from the containment to the environment that occurs before the effective implementation of offsite emergency response and protective actions such that there is a potential for early health effects.

Large Early Release Frequency (LERF)–The expected number of large early releases per unit of time.

Min-Max Method–A method used to determine the HCLPF capacity of an accident sequence from the HCLPF capacities of the contributing SSC failures, or the HCLPF capacity of the plant as a whole from the HCLPF capacities of a group of seismic-initiated accident sequences. The overall HCLPF capacity of two or more SSCs that contribute to a sequence using OR Boolean logic is equal to the lowest individual HCLPF capacity of the constituents of the group. If AND Boolean logic is used, the HCLPF capacity of the group is equal to the highest individual HCLPF capacity of the constituents. When evaluating several accident sequences to determine the "plant level HCLPF capacity," the plant-level HCLPF capacity is equal to the lowest of the sequence-level HCLPF capacities.

Review Level Earthquake (RLE)–A representation of an earthquake ground motion in the form of a response spectrum (applied at a certain depth or location) used as the basis for the analyses performed in a seismic margin assessment. The RLE is also often used as a "figure of merit" for judgments based on the SMA as to the adequacy of the "seismic capacity" of an individual SSC, of an accident sequence, or of the plant as a whole. Specifically, when performing an SMA, an individual SSC's seismic HCLPF capacity (or the capacity of an accident sequence or of the plant as a whole) typically is compared to the RLE. If the HCLPF capacity is greater than the RLE, the inference is that the capacity is "adequate" or that there is "adequate seismic margin." However, this latter judgment of adequacy depends on the application of the SMA results as used by a decision maker. In the EPRI SMA methodology, and in some other early SMA literature, the RLE is known as the "seismic margin earthquake," but these are two names for an essentially identical construct. For the purposes of addressing the 50.54(f) letter, the RLE is the envelope of the safe-shutdown earthquake (SSE) and the GMRS, as discussed in Section 4.3.1. The RLE should be applied at the GMRS location.

Safe Shutdown Earthquake Ground Motion (SSE)−The vibratory ground motion for which certain SSCs are designed, pursuant to Appendix A to 10 CFR Part 100, to remain functional. The SSE for the site is characterized by both horizontal and vertical free-field ground motion response spectra at the free ground surface.

Seismic Equipment List (SEL)–The list of all SSCs that require evaluation in the seismic fragilities task of a seismic margin assessment.

Soil Liquefaction–A fluid-induced loss of soil strength caused by seismic ground motion with two typical failure modes: (1) flow failure where the shear strength of the soil drops below the level needed to maintain stability; and (2) cyclic mobility failure (lateral spread). Either failure mode can lead to excessive strains and displacements that could result in unacceptable performance of supported SSCs.

3.0 Background, Overview, and Issues Related to the Seismic Margin Assessment Method

3.1 Background on Seismic Margin Assessment

A panel of NRC-supported experts initially developed the SMA method between 1984 and 1985 to assess the capability of nuclear power plants to withstand earthquakes above their design basis. The SMA method was first described in NUREG/CR-4334, which has been further supplemented by NUREG/CR-4482, "Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants," and NUREG/CR-5076, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants: The Importance of BWR Plant Systems and Functions to Seismic Margins." Shortly after the NRC's development of the SMA method, the EPRI developed a related but different SMA methodology, as described in EPRI Report NP-6041-SL (EPRI, 1991). These two methods are commonly called the "NRC SMA method" and "EPRI SMA method," respectively.

To assess the methods, researchers conducted a trial of the NRC SMA method at the Maine Yankee Nuclear Power Plant (NUREG/CR-4826, "Seismic Margin Review of the Maine Yankee Atomic Power Station"), a trial of the EPRI SMA method at the Catawba Nuclear Station (EPRI Report NP-6359, "Seismic Margin Assessment of the Catawba Nuclear Station" (EPRI, 1988)), and trials of both methods were conducted concurrently for the Hatch Nuclear Plant

(NUREG/CR-5632, “Seismic Margin Review of Plant Hatch Unit 1: System Analysis”). After these three trial applications, the NRC endorsed using either the NRC or EPRI methods when all of the licensed nuclear power plants undertook the IPEEE program in response to Supplement 4 to NRC Generic Letter 88-20, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities-10 CFR 50.54(f),” issued April 1991. However, in the IPEEE staff guidance in NUREG-1407, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” the NRC specified certain enhancements to the SMA methods if either was to be used to respond to the IPEEE information request.

About two-thirds of the operating plants performed an EPRI SMA to satisfy the IPEEE information request. Two performed an NRC SMA. The rest performed a seismic probabilistic risk assessment (SPRA).

3.2 Comparison with Electric Power Research Institute Success Path-Based Seismic Margin Assessment

The most important technical difference between the NRC SMA methodology and the EPRI SMA methodology is the SMA systems analysis approach. The systems analysis develops the seismic equipment list (SEL) that is the basis for the seismic fragilities part of the SMA methodology. The NRC SMA method, as described in NUREG/CR-4334, uses a seismic PRA fault-tree and event-tree approach to delineate accident sequences, although it is limited to only a selected number of safety functions necessary to prevent core damage. The EPRI SMA method uses a success-path approach in which two success paths are the basis for the systems analysis. This approach defines and evaluates the HCLPF capacity of those SSCs required to bring the plant to a stable condition (hot or cold shutdown) and to maintain that condition for 72 hours.

Many of the other key systems-analysis assumptions are the same in both methods, including assuming that the earthquake always causes unrecoverable loss of offsite power, that only systems and components needed to accomplish certain core-damage-prevention functions are within the scope, and that certain “screening tables” in the guidance reports can be used for screening in or out major SSC categories. These screening tables were developed as the result of expert judgments by the authors of the methodology guidance reports, based in turn on earthquake experience data, test data, and various analyses that then existed in the literature.

Another key difference between the two methods is their methodological guidance for developing the “HCLPF capacity”[1] of an individual SSC. The EPRI method uses the CDFM method for determining the HCLPF capacity, whereas the NRC method permits the CDFM method, but prefers use of the fragility analysis (FA) method that is also known as the “separation of variables” method. The CDFM method directly analyzes for the HCLPF capacity of an individual SSC, whereas the FA method develops a fragility curve from which the HCLPF capacity is extracted.

Part 5 of the ASME/ANS PRA standard, endorsed by the NRC in Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” permits use of the CDFM method to derive the SPRA fragilities for some applications (see note 1 to supporting requirement FR-F1 of the ASME/ANS PRA

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1 See Section 2 of this document for a definition of the high confidence low probability of failure (HCLPF) capacity.

standard). The fragility curve for an SPRA based on the CDFM method would be developed from the HCLPF and assuming a generic composite β, as set forth in that standard.

3.3 Enhancements to the NRC and EPRI Seismic Margin Assessment Methods Required in the IPEEE Program

When the NRC provided guidance in NUREG-1407 for performing the IPEEE, neither the NRC SMA method nor the EPRI SMA method was judged adequate as originally developed. NUREG-1407 specified certain enhancements if an SMA analysis was to be used in the IPEEE. Today, based on experience gained over the two intervening decades, the NRC staff has judged additional enhancements to be necessary if an SMA evaluation is to be used to satisfy the 50.54(f) letter. These various enhancements, over and above what is in the original NRC or EPRI guidance, are summarized in the following sections.

3.4 Features and Enhancements Necessary if an SMA is to be Used to Respond to the NRC March 12, 2012, 50.54(f) Request for Information letter

A list of the high-level features and enhancements necessary if an SMA is to be used for the purposes of responding to the 50.54(f) letter is presented below. Some of these topics are similar to staff positions taken during the IPEEE program; others are additional enhancements. Where appropriate, a staff position further describing the necessary enhancement is presented in Section 4.

- The SMA should use a systems-analysis approach that begins by following the NRC SMA methodology, using event trees and fault trees, with enhancements; an EPRI SMA approach using success-path systems logic is not acceptable.

- The SMA should be a full-scope SMA, not a focused-scope or reduced-scope SMA (as described in NUREG-1407; see Section 4.2.1 for additional details).

- The systems model should be enhanced over what was contained in the original NRC SMA guidance (in NUREG/CR-4334 and NUREG/CR-5076) and the NRC's IPEEE guidance (in NUREG-1407) (see Section 4.4 for more detail).

- The scope should include certain containment functions and containment systems to enable analysis of the plant-level HCLPF for large early release.

- The "mission time" should extend to either 72 hours or when the plant reaches a stable state, whichever is later (see Section 4.4.2).

- The use of the so-called "min-max" method, if selected, should follow certain guidance (see Section 4.6.2). The convolution method is the preferred method.

- When developing sequence-level and plant-level HCLPF capacities, the analysis should differentiate between those sequences that lead to core damage and those that lead to a large early release.

- The HCLPF capacities for sequences with non-seismic failures and human actions and HCLPF capacities for sequences without them should be separately reported.

- The SMA analysis should assume that the earthquake causes an unrecoverable loss of offsite power. (The IPEEE specified the same assumption and is a standard assumption in all SMAs.)

4.0 Staff Positions on Individual Technical Issues

4.1 Introduction

4.1.1 **Organization of the staff positions**

This section provides discussions and NRC staff positions on various technical issues. The topics are broken into several subsections, as shown in the figure below. Additional guidance is provided on the topics shown in the figure. Sections 5 and 6 of this document provide staff positions on peer review and documentation.

SMA Scope 4.2	• Addition of certain containment functions and systems to assess LERF • HCLPF capacities for core-damage and large early release sequences • Separate analysis of HCLPF capacities of sequences with and sequences without non-seismic failures and human errors • Chatter analysis and treatment of high-frequency response of certain SSCs
Ground Motion and In-Structure Reponse 4.3	• Selection of the Review Level Earthquake • Soil failures • Development of in-structure response spectra • Median seismic responses of systems and components
Systems Analysis 4.4	• Enhancements to the PRA-type systems SMA model beyond those in the original guidance • "Mission time" for the accident analysis • Selection of the Seismic Equipment List
Fragility and Capacity 4.5	• Plant walkdown methodology • Screening approach and level for of SSCs • Fragility analysis method for evaluation of the HCLPF capacity of an SSC • CDFM method for evaluation of the HCLPF capacity of an SSC
SMA Integration 4.6	• Plant margin evaluation using the Convolution Method for sequence-level and plant-level HCLPF capacity • Guidance on using the "Min-Max" method for sequence-level and plant-level HCLPF capacity

4.1.2 The Screening, Prioritization, and Implementation Document

NRC staff currently is engaged with industry in developing guidance for some specific technical elements needed to address the 50.54(f) letter. This applicable guidance will be provided in a screening, prioritization, and implementation document (SPID), which has an expected publication date of November 2012. The industry is developing the SPID with a significant level of review and input from NRC staff; and it is anticipated that the SPID will be submitted to the NRC by the Nuclear Energy Institute. Once the NRC staff completes its review of the SPID and determines its adequacy, the NRC staff will endorse the SPID. To the extent appropriate, the applicable SPID positions are incorporated into this ISG document, and Appendix A provides some additional information on SPID positions.

4.2 Seismic Margin Assessment Scope Issues

4.2.1 Introduction

Three methods currently can be used to perform an SMA: the PRA-based SMA method (as described in ISG DC/COL-ISG-020), the NRC SMA method (as described in NUREG/CR-4334, supplemented by NUREG/CR-4482 and NUREG/CR-5076), and the EPRI SMA method (as described in EPRI NP-6041-SL Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" (EPRI, 1991)). The three methods differ in two key areas: the initiators considered in the analysis and the system logic model approach used. As a result, the three methods are appropriate for different applications and objectives.

The PRA-based SMA method and the NRC SMA method both use a fault-tree/event-tree representation of the systems model; although the trees used in the NRC SMA method are simplified compared to the PRA-based method. NUREG/CR-4334 limited the systems considered in the NRC SMA method to those that support frontline functions, which is a significant simplification over the PRA-based SMA method (and a seismic PRA). This ISG enhances the systems considered to include those needed for the recirculation phase and cold shutdown and systems that perform certain accident-mitigation and containment functions. The EPRI method does not use an event-tree or logic-tree approach, but rather defines two success paths that can be used to address a transient and a small loss-of-coolant-accident (LOCA). While the PRA-based SMA method considers all potential initiators, both the NRC and the EPRI SMA methods consider only seismically induced transients and small LOCAs.

This ISG addresses only the NRC SMA method. The EPRI SMA method does not achieve the objectives described in the 50.54(f) letter because of its use of a success path approach (as discussed above). In principle, the full PRA-based SMA can be used as long as the licensee follows the guidance in this ISG, the SPID, and ISG DC/COL-ISG-020 (with the exception of elements that clearly and only relate to new reactor licensing). In addition, ISG DC/COL-ISG-020 does not address some specific considerations pertinent to the 50.54(f) request.

In the IPEEE program, a tiered approach was taken and three "scope" levels were developed for plants to use when taking into account estimates of site-specific hazard levels. The definitions of these levels, termed "reduced scope," "focused scope," and "full scope," are provided in NUREG-1407. These IPEEE-based definitions are not used within this ISG, nor within the recommended program, to address the 50.54(f) letter, except when specifically referring to the IPEEE program or NUREG-1407. The use of the term "full scope" NRC SMA in the context of this ISG and the 50.54(f) letter implies use of the NRC SMA with the enhancements described in this document.

The NRC staff will determine the applicability of the SMA method to address the 50.54(f) letter for each site during the screening and prioritization phase of the seismic reevaluation process. The approaches described in this ISG are to be applied by all plants conducting a SMA in response to the 50.54(f) letter. Plants for which an SMA is found to be appropriate may choose to conduct a seismic PRA instead of an NRC SMA.

4.2.2 Addition of certain containment functions and containment systems to include assessment of large early release

Technical Issue: To understand the potential for large early radioactive releases, the scope of the NRC SMA analysis as described in NUREG/CR-4334 needs to be extended to evaluate accident sequences beyond the “early” “preventive” safety functions in the original scope. Specifically, certain containment functions and systems are to be included in the SMA’s scope.

Staff Position: The SMA’s scope should be extended to include assessment of large early release. Certain containment functions and containment systems needed to address large early release are discussed in Section 4.4.1.

4.2.3 Differentiation between HCLPF capacities for core-damage sequences and for large early release sequences

Technical Issue: The 50.54(f) letter requires that information about sequences leading to core damage and information about sequences leading to large radioactive releases are both developed and reported.

Staff Position: When analyzing for sequence-level and plant-level HCLPF capacities, the SMA analysis should separately determine the HCLPF capacities for the core-damage endpoint and for the large-early release endpoint.

4.2.4 Separate analysis of HCLPF capacities of sequences *with* and sequences *without* non-seismic failures and human errors

Technical Issue: In a typical seismic PRA, an important fraction of all of the accident sequences involve a combination of failures caused by the earthquake and other failures not related to the strong motion. Therefore, the SMA analyst is required to separately determine the HCLPF for the accident sequences containing only seismic failures and the HCLPF for the accident sequences containing both seismic and non-seismic failures.

Neither non-seismic failures nor human errors are explicitly accounted for using the “min-max” method in a traditional SMA when the HCLPF capacity for an individual accident sequence is developed; nor are they explicitly accounted for in aggregating to a “plant-level HCLPF capacity.” The convolution method is one way to include non-seismic factors in the quantification (see Section 4.6.1).

Staff Position: When developing sequence-level and plant-level HCLPF capacities, non-seismic failures and human errors should be included. The analysis should separately determine the HCLPF capacities of sequences *with* and sequences *without* non-seismic failures and human errors.

4.2.5 Relay chatter analysis and treatment of high-frequency response of certain SSCs

Technical Issue: The analysis of the chatter of relays during earthquakes has long been a technical concern. NUREG-1407 provided previous guidance for the IPEEE program for full scope plants. The ASME-ANS PRA standard in Part 5 and Part 10 also provides guidance. A related recent concern is whether there are any other SSCs besides relays that are sensitive to high frequency motions.

To address this long-standing question and to facilitate a more consistent response to the 50.54(f) letter, industry has initiated a testing program of potentially high frequency sensitive components that will serve as the technical basis for new guidance. Industry representatives and NRC staff are working jointly on the development of the testing program, which is to be conducted in two phases.

The results of the testing program will be used to confirm the adequacy of equipment response to high frequency input motions for plants for which the GMRS exceeds the safe-shutdown earthquake (SSE) ground motion in the high frequency range (>10hz). Plants that have exceedance only in the high frequency range would screen out from performing an SMA or SPRA, but would still have to address the performance of high frequency components. The testing program will address the performance of potentially high frequency sensitive components across the industry.

The testing program also will provide additional guidance and component capacity information useful to plants undertaking further risk evaluations because they have exceedances in both the high and low frequency ranges. The results will be incorporated into the SPID as additional guidance for addressing the 50.54(f) letter. Once the guidance in the SPID is reviewed and endorsed by NRC staff, it can be used to address the 50.54(f) letter.

Because Phase 2 of the program will finish after publication of the SPID, a final report with additional guidance also will be issued. Once the guidance in the final high frequency testing report is reviewed and endorsed by NRC staff, it can be used to address the 50.54(f) letter as well.

Staff Position: This technical topic is covered in a separate high frequency testing program. Once the NRC endorses the related guidance documents, they can be used to address the 50.54(f) letter.

4.3 SMA Hazard, Ground Motion, and In-Structure Motion Issues

4.3.1 Selection of the Review Level Earthquake

Technical Issue: The SMA methodology uses a review level earthquake (RLE)[2] as the ground motion level used in the analysis. The RLE is a representation of an earthquake ground motion in the form of a response spectrum (applied at a certain depth or location) that is used as the basis for the analyses performed in a SMA.

While the same term was used in the IPEEE program, the method used to determine the RLE for responding to the 50.54(f) letter is different. The 50.54(f) letter specifies that the licensee compare the site-specific GMRS[3] with the plant's SSE ground motion response spectrum and use the

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[2] In some documents, the term "seismic margin earthquake" or SME may be used.
[3] Acceptable methods for development of the GMRS are described in the 50.54(f) letter.

envelope of these two spectra as the RLE. Although the method for developing the RLE differs from earlier studies (such as the IPEEE), its use within the SMA is similar.

Staff Position: The RLE to be used in the SMA for a particular plant is the envelope of the SSE and the GMRS over the entire frequency range. The method for determining the RLE is specified in the 50.54(f) letter. The RLE should be applied at the GMRS location.

4.3.2 Soil failures

Technical Issue: Soil failure analyses include an evaluation for instability, excessive settlement, and liquefaction. EPRI NP-6041 contains guidance on performing these analyses. Fragility for seismically induced liquefaction can be developed using the method described in Appendix G of EPRI 1002988 report, “Seismic Fragility Application Guide” (EPRI, 2002). In this method, the limit state may be defined in terms of the consequences of liquefaction induced settlement on the site configuration of safety-related SSCs, including site layout, umbilical between structures, and buried pipes and concrete electrical ducts when adequate justifications are provided. Additional guidance is found in the ASME/ANS PRA standard.

Staff Position: The assessment should include appropriate soil failure modes using EPRI NP-6041 and EPRI 1002988. Additional guidance is found in the ASME/ANS PRA standard. A more detailed evaluation of plant site conditions using state-of-the-art approaches should be performed if soil failure is deemed to have a significant potential.

4.3.3 Development of In-Structure Response Spectra

Technical Issue: The assessment of structural response and the resulting in-structure response spectra (ISRS) are important aspects of an SMA. Both the overall amplitude and the shape of the response spectra used in the SMA have a significant effect on the SMA results. Therefore, the ISRS must be sufficiently accurate to provide confidence in the SMA results, in terms of the core damage frequency, large early release frequency, and dominant risk contributors identified.

At the same time, a significant amount of existing structural response information in the form of structural models and ISRS (either from the original design, IPEEE, or A-46 programs) is available for operating reactors. Unfortunately, this information represents a wide range of vintages, and the methods used to develop the information vary in their consistency with current accepted practice. Use of this existing structural response information, where appropriate, represents one avenue for reducing the overall level of effort required; however, criteria to determine the continued appropriateness of the models and information must be applied to ensure that the objectives of the 50.54(f) letter are achieved.

In an effort to appropriately optimize the use of existing information, the NRC and industry experts have been working to develop guidance on several aspects of structural response that are included in the SPID and this ISG. Appendix A discusses this guidance in detail. The topics addressed are threefold:

- **Attributes of existing structural models needed for appropriately addressing the 50.54(f) letter.** As described in Appendix A, existing structural "stick" models may be appropriate if they have sufficient complexity and attributes to provide the appropriate level of accuracy needed for the SMA (or PRA). Appendix A provides specific criteria. Models not currently meeting the criteria may be updated.

- **Scaling of ISRS.** The scaling of ISRS is permitted, provided that the spectral shapes of the original input motions and the new RLE are similar and the use of scaling is documented and justified. Scaling of dissimilar spectra is not permitted.

- **Use of fixed base models for soft rock conditions.** The use of fixed base models for structures founded on rock with shear wave velocity greater than 3,500 feet/second is permitted, provided that the general guidance in Appendix A and in the Staff Position below is followed.

An experienced structural engineer should review the use of any existing models or data, which also should be subject to peer review. Use of existing information should be documented in the submission to the NRC and should be adequately justified. The documentation and justification provided should address any potential issues or deficiencies.

Staff Position: Realistic ISRS should be calculated using the guidance on the use of existing information and models provided in the SPID (with any exception provided in the NRC endorsement of the SPID). Appendix A discusses the SPID contents and guidance as it relates to development of the ISRS. If an existing structural model is used, its attributes should be compared to the SPID criteria and its applicability documented and justified. If ISRS scaling is used, it should be consistent with current accepted practice and SPID guidance on the use of scaling. The use of scaling should be documented and justified. The SPID provides the technical basis developed to support the use of fixed base models for structures founded on rock with a shear wave velocity greater than 5,000 feet/second. However, the use of fixed base models on softer rock may be justified in some cases. The use of fixed base models for Vs>3,500 feet/second, if used, should be justified.

An experienced structural engineer should review the use of any existing models or data, which also should be subject to peer review. Use of existing information should be documented in the submission to the NRC and should be adequately justified. The documentation and justification provided should address any potential issues or deficiencies.

4.3.4 Median seismic responses of systems and components

Technical Issue: The system and component seismic responses should be median-centered and based on current state-of-the-art or models and assessment methods that meet the staff position in Section 4.3.3.

Staff Position: Realistic equipment response should be calculated using ASME/ANS PRA Standard Part 10 (High Level Requirement HLR-SM-C) or using the staff position in Section 4.3.3.

4.4 SMA Systems Analysis Issues

4.4.1 Enhancements to the PRA-type systems SMA model beyond those in the original guidance

Technical Issue: The original NRC guidance for the systems-analysis aspect of an SMA analysis is described in NUREG/CR-4334, supplemented by NUREG/CR-4482 and NUREG/CR-5076. The NRC staff provided certain enhancements in the IPEEE guidance (NUREG-1407). For an SMA performed to address the 50.54(f) letter, the scope must be extended to identify sequences involving a potential large early radioactivity release. Certain features and enhancements are considered necessary to accomplish this expanded scope. The staff position describes these features and enhancements.

Staff Position: The staff position is as follows:

Initiating events: The postulated seismic-caused initiating events should include unrecoverable loss of offsite power, small LOCAs, and certain transients, such as safety-relief-valve initiators. Seismic-initiated large LOCAs need not be included.

Safety functions: The original SMA methodology requires consideration of only the “Group A” safety functions as explained in NUREG/CR-4334 for pressurized-water reactors (PWRs) and NUREG/CR-5076 for boiling-water reactors (BWRs). For both PWRs and BWRs, these include subcriticality and early emergency core cooling system injection until stabilization of temperature and pressure. These should be supplemented by systems necessary to achieve core cooling and long-term heat removal for times beyond the “early” period. Depending on the specific design, this can mean including systems and functions through the recirculation phase for a PWR, or through switchover to suppression pool cooling for a BWR, and then establishment of residual heat removal and other functions needed to bring the plant to a stable state.

Scope of containment analysis: As identified in Section 4.2.2, the systems-analysis scope should include certain containment functions and containment systems to address large early releases. Examples include containment penetrations and containment isolation systems, containment pressure suppression and overpressure-protection systems, containment heat removal systems (early and late), and hydrogen control systems. These are needed so that the SMA can differentiate between those rare seismic-initiated core damage accident sequences that lead to large early releases and the larger number of sequences that do not.

Assessment of containment structural failure is outside the scope of an SMA, as set forth in the original guidance.

<u>Successes</u>: The systems model should retain “successes” of those SSCs for which detailed fragility calculations are not required because of their strong seismic capacity. This should be done to facilitate review of the accident sequences. For these SSCs, it is acceptable to use an estimated or generic HCLPF capacity. Section 4.5.2 provides further guidance.

4.4.2 “Mission time” for the accident analysis: until a stable state is reached

Technical Issue: The seismic risk assessment required under the 50.54(f) letter seeks to understand risk contributors for accident sequences that involve the potential for core damage, or that involve the potential for a large release of radioactivity. To understand the latter type of sequence, the SMA analysis should study sequences for as long after the earthquake as is necessary for the reactor to reach a stable state. That stable state might be a “safe” state, or a state involving extensive damage to the core, or a state involving a large radioactivity release, or somewhere in between. The scope of this type of analysis is greater than that for an SMA performed under the IPEEE guidance (NUREG-1407), which calls for a 72-hour mission time as the appropriate scope.

Staff Position: For each potential accident sequence, the mission time for the safety systems and functions that the SMA analysis evaluates should extend either to 72 hours or to the time required to achieve a stable state, whichever is longer.

4.4.3 Plant system and accident sequence analysis: Selection of the Seismic Equipment List

Technical Issue: The SMA systems analysis involves, in part, the selection of an SEL. This is a standard aspect of both SMA and seismic PRA. In addressing the 50.54(f) letter, the SEL should include equipment needed to asses both core damage and large early release. Some past guidance has differed on the inclusion of large early release.

Staff Position: The SEL should include the systems necessary to achieve cold shutdown and the appropriate containment systems, as set forth in Section 4.4.1 above. The starting point for constructing the SEL is the set of SSCs included in the internal events PRA model, to which a number of SSCs with earthquake-specific issues must be added, such as passive components not present in the internal events model whose seismic failure could be important to core damage or large early release.

4.5 SMA Fragility and Capacity Issues

4.5.1 Plant walkdown methodology

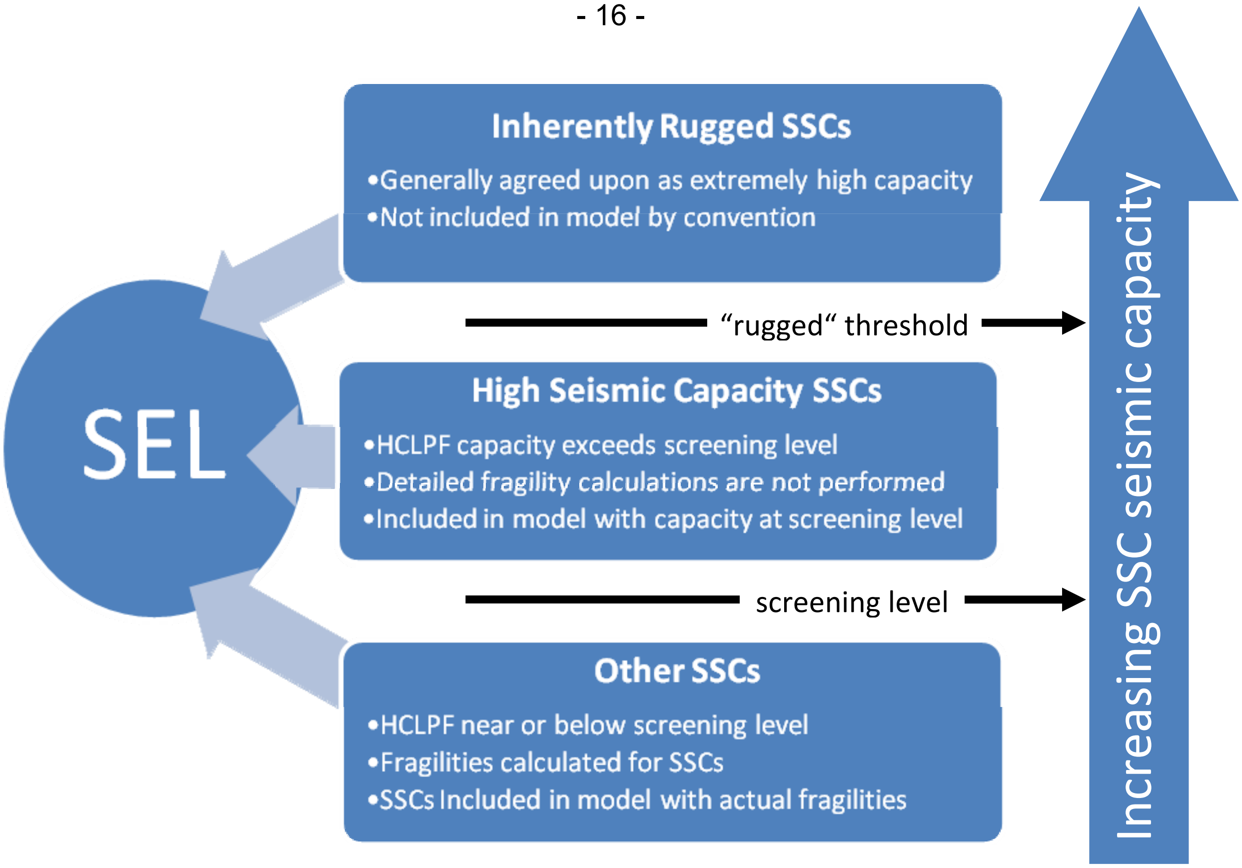
Technical Issue: The seismic walkdown is an important activity in any SMA. The walkdown ensures that the seismic fragilities or margins are realistic and plant-specific, and finds any as-designed, as-built, and as-operated seismic vulnerabilities in the plant. It should be done in sufficient detail and documentation so that the subsequent screening or margin evaluation is traceable. EPRI NP-6041 and NUREG/CR-4334 provide licensee guidance on walkdown.

Technical guidance on the walkdown is found in EPRI NP-6041 and in Part 10 of ASME/ANS PRA Standard, which refers back to EPRI NP-6041.

Staff Position: The seismic walkdown should be conducted in conformance with either Part 10 of ASME/ANS PRA Standard or EPRI NP-6041.

4.5.2 Screening approach and level for SSCs

Technical Issue: The SEL SSCs generally can be broken into three categories or “bins” based on their seismic capacity as compared to the RLE (and in-structure response spectra) at the site of interest. These categories are “inherently rugged” SSCs, “high seismic capacity” SSCs, and other SSCs (i.e., which cannot be shown to have high seismic capacity such that they can be screened out), as shown schematically in the figure below. SSCs that fall into each of these bins are addressed differently within an SMA both in the approach used to determine their capacity and their treatment in the modeling process. By identifying components that are inherently rugged or have high seismic capacity, attention and resources can be focused on determining the capacity of SSCs with a greater likelihood of being risk significant.



Inherently rugged SSCs are those extremely robust components believed to have a seismic capacity beyond any realistic earthquake loading levels (e.g., manual valves). These components generally are mutually agreed upon by the technical community and, by convention, they are not included in SMA or SPRA models. EPRI NP-6041-SL provides guidance on inherently rugged components.

Other SSCs may be less rugged but still have sufficient capacity so that their failures would be unlikely to contribute significantly to the seismic core damage in a seismic SMA. These components, noted as high seismic capacity SSCs, should still be incorporated into the model, although detailed fragility calculations are not warranted. The screening Tables 2-3 and 2-4 in EPRI NP-6041 and the capacity information in EPRI TR-103959, "Methodology for Developing Seismic Fragilities," and EPRI 1002988 are helpful for assessing if a particular SSC has a seismic capacity (HCLPF) higher than the screening level (which is generically higher than the RLE.) Other sources of information, such as design drawings and IPEEE analyses also may be useful. For the components that can be classified as having high seismic capacity, detailed margin evaluations need not be performed (except for anchorage calculations), but the SSCs must be included in the systems model (both the event trees and the fault trees) to assist the analyst in understanding the most risk-significant accident sequences. The HCLPF capacities assigned are set equal to the screening level. The design review and walkdown should confirm the validity of this assignment.

Seismic fragilities must be calculated for all SSCs that are neither inherently rugged nor shown to have high seismic capacity. These SSCs are included in the model with their actual capacities determined using the methods described in Sections 4.5.3 and 4.5.4 of this document.

In the past, different screening approaches have been applied in SMA. This experience has shown that the ability to obtain risk insights from an SMA can be significantly curtailed if the screening level is set too low. An example was the IPEEE program, in which a number of plants found that the "surrogate elements" (a stand-in for screened components) dominated the accident sequences. This outcome greatly limited the ability to identify the actual most risk significant SSCs.

These past lessons indicate that additional guidance must be provided to ensure that the objectives of the 50.54(f) letter are met. As a result, the NRC and industry experts have developed guidance on the appropriate screening approaches and levels to be used for inclusion in the SPID. That guidance is summarized below.

The risk information needed includes core damage frequency, large early release frequency, and the identification of the dominant risk contributors, as indicated in the March 12, 2012, letter. To gain the necessary risk insights from an SMA, the screening level must be set well above the RLE. Based on the results of the analyses performed to support the guidance in the SPID, either of two criteria may be used for the initial screening of SSCs.

The screening level may be set as either:

- A screening level consistent with an HCLPF capacity that is 2.5 times the RLE, or
- A screening level equivalent to the HCLPF that leads to a frequency of failure on the order of $5x10^{-7}$/yr using a mean point estimate approach, an assumed composite β_c, and the site hazard.

A selected screening level based on one of the above criteria can be used in reviewing previous IPEEE, A-46, or design basis calculations to judge if explicit fragility or HCLPF calculations are needed for each SSC.

The NUREG/CR-4334 and EPRI-NP-6041 screening tables can be used to identify and assign conservative HCLPF values to the high seismic capacity SSCs using a screening level that is higher than the RLE as defined above. The use of these tables must include satisfying caveats associated with the tables as well as anchorage evaluations, as appropriate. This enhancement is from earlier guidance, and the IPEEE and NUREG/CR-4334, which allowed the screening level to be set at the RLE.

Once the SMA analysis has been performed, a check must be conducted to ensure that none of the components identified as among the dominant contributors to either the HCLPF capacity for core damage or the HCLPF capacity for large early release are high seismic capacity SSCs. If any high seismic capacity SSCs are identified as among the dominant contributors, then the actual HCLPF capacities of the components must be developed and the analysis rerun.

A check should also be performed to ensure that each of the non-seismic failures associated with high seismic capacity SSCs are retained in the model.

Staff Position: When identification of high-seismic capacity components is performed, the basis for identifying them, including supporting documents, should be fully described. EPRI NP-6041-SL Revision 1 and NUREG/CR-4334 guidance may be used, provided the following enhancements are applied. (Use of screening tables of NUREG/CR-4334 and EPRI NP-6041 must include satisfying caveats associated with the tables and should include anchorage evaluations, as appropriate.)

- The components identified as having a high seismic capacity as compared to the screening level should be assigned capacities equal to the screening level unless a detailed fragility calculation has been performed. These components should be retained in the system model for accident sequence analysis.

- The screening level may be set as either:

    - A screening level consistent with an HCLPF capacity that is 2.5 times the RLE, or

    - A screening level equivalent to the HCLPF that leads to a frequency of failure on the order of 5x10-7/yr using a mean point estimate approach, an assumed composite variability, and the site hazard.

- Once the SMA analysis has been performed, a check must be conducted to ensure that none of the following conditions exist:

    - A high seismic capacity SSC that was assigned a seismic capacity equal to the screening level is identified as a dominant contributor to HCLPF for core damage.

    - A high seismic capacity SSC that was assigned a seismic capacity equal to the screening level is identified as a dominant contributor to HCLPF of large early release.

If any of the above conditions exist, the screening level should be reevaluated and adjusted and actual HCLPF capacities of the components should be analyzed using the methods described in

Sections 4.5.3 or 4.5.4, and incorporated into the model to the extent necessary, and the analysis should be rerun.

4.5.3 Fragility analysis (FA) method for evaluation of the HCLPF capacity of an SSC

Technical Issue: The FA method is described in a number of references (e.g., NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," NUREG/CR-4334, NUREG/CR-4482, and EPRI TR-103959). Typically, the seismic fragility of a component is characterized by a double lognormal model whose parameters are A_m, β_R, and β_U. A_m is the median capacity. β_R is the logarithmic standard deviation of the capacity and represents the variability due to the randomness of the earthquake characteristics for the same acceleration and to the structural response parameters that relate to these characteristics. β_U is the logarithmic standard deviation of the median capacity and represents the uncertainties in models and model parameters. The "seismic margin" is defined in terms of the HCLPF capacity and is calculated using the equation

$$\text{HCLPF} = A_m \exp[-1.64*(\beta_R + \beta_U)]$$

For some applications, it may be sufficient to develop a mean fragility curve characterized by a lognormal probability distribution with parameters of A_m and β_c, where $\beta_c = (\beta^2_R + \beta^2_U)^{½}$ is the logarithmic standard deviation of composite variability. The HCLPF capacity is taken as the 1 percent conditional-probability-of-failure value, i.e.,

$$\text{HCLPF} = A_m \exp(-2.33\ \beta_c).$$

For every component in the plant system model, the FA method evaluates the family of fragility curves or a single composite fragility curve from which the HCLPF capacity (known as seismic margin) is estimated.

As an alternate approach, the CDFM method (described below) can be used with a generic β_c to develop the fragility of an SSC. The use of generic β_c, along with recommended values, is found in the SPID (upon SPID endorsement by the staff).

Staff Position: Fragility analysis for SSCs should be performed either in accordance with Part 5 of the ASME/ANS PRA standard or using the CDFM method with a generic β_c following the guidance in the SPID and Sections 4.5.4 and 4.6.1 below.

4.5.4 CDFM method for evaluation of the HCLPF Capacity of an SSC

Technical Issue: The FA method of estimating the HCLPF capacity of SSCs, although universally applicable, does require the median factors of safety for different variables affecting the response and capacity to be estimated as well as their logarithmic standard deviations. "Seismic margin" for any SSC is defined in terms of the HCLPF capacity. The HCLPF capacity of an SSC can be calculated directly using the CDFM method. EPRI NP-6041-SL (EPRI, 1991) describes the CDFM method and provides several examples. In this procedure, the values of different variables that figure into the HCLPF capacity evaluation are judiciously selected at median values (best estimate or structural model and conservative estimate of median damping) or some conservative values (e.g., code specified minimum material strength or 95 percent exceedance actual strength if test data are available; 84 percent non-exceedance ground response spectrum). The HCLPF calculation of capacity follows the deterministic procedures used in the seismic design and qualification of SSCs.

Staff Position: If the HCLPF capacity is evaluated using the CDFM method, the analysis should be performed in accordance with the EPRI NP-6041-SL, with any necessary adjustments as discussed in EPRI TR-103959 (EPRI, 1994) and EPRI 1019200 (EPRI, 2009).

4.6 SMA Integration Issues

4.6.1 Sequence-level and plant-level HCLPF capacity: plant margin evaluation using the convolution method

Technical Issue: The convolution method for evaluating plant margin is preferred over the “min-max” method described in Section 4.6.2. In the convolution method, accident sequences are evaluated by combining input fragility curves according to the Boolean expression for each sequence. Seismic and random or human failure probabilities are calculated and combined (convolved) for discrete intervals of ground acceleration and then integrated over the range of interest. It is also important to keep the “success” events (i.e., the “PRA event” when a component does not fail) in the calculation. The result is the family of fragility curves for each accident sequence. By combining these accident sequences that result in core damage or large early release, the plant level fragility curves are obtained. The “plant-level seismic margin” is then evaluated as the HCLPF capacity, defined as equal to the seismic ground acceleration at which the probability of failure is equal to 5 percent with a confidence level of 95 percent. This method of developing the plant level fragility curves will retain the information for the analyst and the NRC to develop accident sequence frequencies, CDF and LERF calculations using the site-specific seismic hazard as needed. The method is described in Appendix 5-A of the ASME/ANS PRA standard. If a single composite fragility curve is input for each SSC in the accident sequences, the resulting plant level fragility curve also will be a single curve and the plant seismic margin will be the HCLPF capacity, defined in this case as the ground acceleration at which the probability of failure is equal to 1 percent.

To simplify both the seismic PRA and SMA analyses, a hybrid method suggested in EPRI Report TR-103959 (EPRI, 1993) and Kennedy (1999) could be used. The main feature of this method is the development of a seismic fragility starting with the HCLPF capacity. First, the HCLPF capacity of the component is estimated using the CDFM method. Next, the logarithmic standard deviation β_c is estimated using judgment and following the guidance given in Kennedy (1999). For structures and major passive mechanical components mounted on ground or at low elevations within structures, β_c typically ranges from 0.3 to 0.5. For active components mounted at high elevations in structures the typical β_c range is 0.4 to 0.6. When specific information is not available, values of β_c as provided in the SPID are recommended (upon SPID endorsement by the staff). The median capacity is calculated using the equation

$$A_m = HCLPF * \exp(2.33\ \beta_c)$$

and an approximate fragility curve for the component is thereby obtained. Using this composite fragility curve for each component in the system model, the plant level fragility curve is obtained following the convolution approach described above. Reed and Kennedy (1994) further recommend that this approximate fragility curve be used for each component in the systems analysis to identify the dominant contributors to the seismic risk (e.g., core damage frequency). For a few components that dominate the seismic risk, more accurate fragility parameter values should be obtained and a new quantification done to obtain a more accurate mean core damage frequency, and to confirm that the dominant contributors have not changed.

Staff Position: The convolution method for evaluating plant margin is preferred over the "min-max" method described in Section 4.6.2. If the convolution method is used, the analyst should perform a margin evaluation of accident sequences using the composite fragility curves for SSCs, mean values of random unavailabilities, and operator error rates. The analysis should also take into account the success terms and include the SSCs that have been assigned HCLPF capacities equal to screening level (and larger than RLE). For the important accident sequences (i.e., direct core damage, large early release, and low seismic margins), the full family of fragility curves (A_m, $\beta_{R,}$ and β_U) and probability distributions on random failure rates and operator error rates should be used in the convolution procedure to obtain the accident sequence fragility curves and plant level fragility curves. The seismic margin of the plant is then evaluated as the plant-level HCLPF capacity, defined as the ground acceleration value corresponding to the 95 percent confidence of not more than 5 percent probability of failure.

4.6.2 Sequence-level and plant-level HCLPF capacity: guidance on using the "min-max" method

Technical Issue: The min-max method (see the definition in Section 2) is a way that an SMA analysis can derive an approximate <u>sequence-level HCLPF capacity</u> from the HCLPF capacities of individual SSCs that comprise the accident sequence, or derive an approximate <u>plant-level HCLPF capacity</u> from the most important (lowest) sequence-level HCLPF capacities that emerge from the SMA analysis.

However, the min-max method sometimes is only a rough approximation and it gives an HCLPF capacity that can be either higher or lower than the "true" value derived from a full seismic PRA that does not make this approximation. For the case of a single "accident sequence," the discrepancy in the reported sequence-level HCLPF capacity arises when two (or more) HCLPF values for individual SSCs are close numerically. The distortion can be either conservative (too low) or non-conservative (too high), depending both on the AND-OR structure of the sequence's Boolean logic and on whether the actual (unknown) fragility curves for these SSCs have steep or shallow shapes (small or large β_c values.) The plant-level HCLPF capacity can be similarly distorted.

The existence of this potential distortion requires the SMA analyst to be alert to the problem. In cases like those mentioned, the SMA analyst should derive the sequence-level HCLPF capacity using the convolution approach.

The distortion is not important, however, in two cases: if a single SSC "dominates" the HCLPF capacity of an accident sequence or if a single accident sequence "dominates" the overall seismic risk profile. The analyst must justify if the min-max approach is used.

Staff Position: If the min-max approach is used, it should be done in accordance with NUREG/CR-4334 and a justification should be provided that this approach provides reasonable estimates for the sequences under consideration and at the plant level. The convolution approach (discussed in Section 4.6.1) is the preferred approach.

5.0 <u>Staff Positions on Documentation</u>

5.1 Documentation Content

Technical Issue: To address the request in the 50.54(f) letter, the SMA should be conducted and documented in accordance with the provisions in this ISG and the 50.54(f) letter. Documentation to be submitted to the NRC includes a number of elements, as shown below.

Staff Position: For plants that perform an SMA, the following information is requested (in each of the following elements a description of how the applicable ISG positions are met should be included. Any alternate approach should be clearly identified along with its technical basis):

(1) Describe how the RLE was developed and used in the SMA and the location at which the RLE was applied (i.e., the control point elevation)

(2) The definition and justification of the response spectrum shape used for the FA of SSCs, accident sequences, and the plant, if it differed from the RLE

(3) A summary of the plant system models, including event trees and fault trees and how they were developed

(4) A description of the methodologies used to quantify the seismic margins of high confidence of low probability of failure (HCLPF) capacities of SSCs, together with key assumptions. This should include details of response analysis, generation of ISRS, and other details

(5) A detailed list of the SSC seismic margin values with reference to the method of seismic qualification, the dominant failure modes, and how the margin analysis is principally supported (e.g., analysis, test data, experience data)

(6) For each analyzed SSC, the parameter values defining the seismic margin (e.g., the HCLPF capacity and any other parameter values such as the median acceleration capacity and the logarithmic standard deviation or “beta” values) and the technical bases for the values

(7) The general bases for screening SSCs, including screening levels and lists of SSCs considered inherently rugged and a list of SSCs considered as high capacity SSCs

(8) Identification of the methods used to calculate sequence-level and plant-level HCLPFs

(9) Risk-significant sequences, dominant cut-sets, and associated Booleans for both core damage and large early release

(10) Sequence-level and plant level HCLPF capacities for both core damage and large early release

(11) A discussion of sensitivity to random failures and operator errors

(12) A discussion of the treatment of uncertainties

(13) A discussion of how the dominant sequences are identified

(14) A description of the process used to ensure that the SMA is technically adequate, including a description of the approach to peer review, the dates and findings of peer reviews, and a description of how peer review findings were closed out

(15) A list of the identified plant-specific vulnerabilities and actions planned or taken

5.2 Separate Reporting of HCLPF Capacities of Dominant Sequences for Core Damage and for Large Early Release

Technical Issue: See the issue discussed in Section 4.2.3. This is the documentation requirement.

Staff Position: When reporting sequence-level and plant-level HCLPF capacities, the SMA analysis should separately report HCLPF capacities for the core-damage endpoint and the large-release endpoint.

5.3 Separate Reporting of HLCPF Capacities of Sequences *with* and Sequences *without* Non-Seismic Failures and Human Errors

Technical Issue: See the issue discussed in Section 4.2.4. This is the documentation requirement.

Staff Position: When reporting sequence-level and plant-level HCLPF capacities, the SMA analysis should separately report HCLPF capacities for sequences *with* and sequences *without* non-seismic failures and human errors.

5.4 Information Retained for Audit

Technical Issue: Some additional information, beyond that submitted to the NRC in response to the 50.54(f) letter, should be retained for NRC audit. This is for both reviewing the response to the 50.54(f) letter, and for any future uses of the program analyses and results.

Staff Position: The information retained for NRC audit should include (but is not limited to):

- applicable event trees and fault trees
- current versions of the system notebooks (if applicable)
- walkdown checklists and reports
- evaluation results

In general, all documents essential for a practitioner in the field to understand and trace what was done in the SMA should be retained. In addition, the way in which the validity of these documents has been ensured should be documented.

6.0 Staff Positions on Peer Review Attributes, Activities, and Documentation

Technical Issue: The peer review is a key element of the SMA process that increases confidence and assurance that the results of the assessment are reliable and provide the information necessary for regulatory decisions. Appropriate documentation of the peer review process is also important.

Staff Position:[4] **Peer review should include the following attributes:**

- The peer review process should be consistent with the comments provided in the NRC staff letter for NEI 12-13, "External Hazards

[4] The staff positions below are consistent with the peer review process set forth in the ASME/ANS PRA standard, as endorsed in Regulatory Guide 1.200.

PRA Peer Review Process Guidelines,"[5] as well as Regulatory Guide 1.200.

- An in-process peer review, as described in the draft ISG on NEI 12-13 "External Hazards PRA Peer Review Process Guidelines," is preferred over a one-time late stage review, but the latter is also acceptable.

- Peer reviewers on various technical elements should have the opportunity to interact with each other when performing the reviews, and on critical items (e.g., results of screening or the SEL development), the peer review should be conducted as a team.

- Particular attention should be paid to justifications for use of models or methods that are not consistent with current practice (e.g., the use of the original structural models, the site response assessment using limited data). In cases in which the SPID is used, reviews should be based on the adherence to the approach and intent of the SPID guidance and the adequacy of the model to address the 50.54(f) letter.

- The peer review process includes a review of the following SMA activities:

    - selection of the SSCs included on the SEL

    - review a sample of the documentation from the seismic walkdowns

    - seismic response analyses

    - seismic HCLPF capacity assessments for individual SSCs

    - sequence-level and plant-level HCLPF quantification

    - final report

The peer review team should be assembled based on the following considerations:

- The peer review team should have combined experience in the areas of systems engineering, seismic capability engineering, and seismic PRAs or seismic margin methodologies.

- The reviewer(s) focusing on the seismic fragility work should have successfully completed the Seismic Qualification Utility Group (SQUG) Walkdown Screening and Seismic Evaluation Training Course or equivalent, or shall have demonstrated experience in seismic walkdowns.

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5 The NRC letter with comments on NEI 12-13 can be found at ADAMS Accession No. ML12321A280.

- One of the peer reviewers should be designated as the overall team leader. The peer review team leader is responsible for the entire peer review process, including completion of the final peer review documentation. The team leader is expected to provide oversight related to both the process and technical aspects of the peer review. The team leader also should pay attention to potential issues that could occur at the interface between various activities.

- Reviewers should be independent of those who are doing the work.

The peer review process should be clearly documented in the report submitted to the NRC. Documentation in the report should include the following:

- The names and qualifications of the team members.

- A description of the peer review process.

- A discussion of the key findings and a discussion on how the findings were addressed.

- Information on the disposition of comments.

- The review of the final report.

- The conclusions of the peer review.

- The peer review report should be documented in a separate report.

7.0 Citations

American Society of Mechanical Engineers/American Nuclear Society, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Standard ASME/ANS RA-Sa-2009, 2009.

Electric Power Research Institute, "Seismic Margin Assessment of the Catawba Nuclear Station," EPRI Report NP-6359, Palo Alto, California, 1988.

Electric Power Research Institute, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," EPRI Report NP-6041-SL, Revision 1, Palo Alto, California, 1991.

Electric Power Research Institute, "Methodology for Developing Seismic Fragilities," EPRI Report TR-103959, Palo Alto, California, 1994.

Electric Power Research Institute, "Seismic Fragility Application Guide," EPRI Report 1002988, Final Report, Palo Alto, California, December 2002.

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Kennedy, R.P., “Overview of Methods for Seismic PRA and Margins Methods Including Recent Innovations,” in *Proceedings of the OECD/Nuclear Energy Agency Workshop on Seismic Risk*, Tokyo, Japan; August 10-12, 1999 (available from the OECD Nuclear Energy Agency, Le Seine St.-Germain, 12 boulevard des Iles, F-92130 Issy-les-Moulineaux, France).

U.S. Nuclear Regulatory Commission, “Seismic Margin Review of Plant Hatch Unit 1: System Analysis,” NUREG/CR-5632, 1990.

U.S. Nuclear Regulatory Commission, “PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants,” NUREG/CR-2300, December 1982.

U.S. Nuclear Regulatory Commission, “An Approach to the Quantification of Seismic Margins in Nuclear Power Plants,” NUREG/CR-4334, August 1985 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090500182).

U.S. Nuclear Regulatory Commission, “Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants,” NUREG/CR-4482, 1986 (ADAMS Accession No. ML12069A017).

U.S. Nuclear Regulatory Commission, “Seismic Margin Review of the Maine Yankee Atomic Power Station,” NUREG/CR-4826, in 3 vols. 1987.

U.S. Nuclear Regulatory Commission, “An Approach to the Quantification of Seismic Margins in Nuclear Power Plants: The Importance of BWR Plant Systems and Functions to Seismic Margins,” NUREG/CR-5076, 1988.

U.S. Nuclear Regulatory Commission, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities- 10 CFR 50.54(f),” Generic Letter 88-20, Supplement No. 4, April 1991. Available at http://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1988/gl88020s4.html.

U.S. Nuclear Regulatory Commission, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” NUREG-1407, June 1991 (ADAMS Accession No. ML063550238).

U.S. Nuclear Regulatory Commission, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” Regulatory Guide 1.208, 2007 (ADAMS Accession No. ML070310619).

U.S. Nuclear Regulatory Commission, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Regulatory Guide 1.200 Revision 2, March 2009 (ADAMS Accession No. ML090410014).

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

U.S. Nuclear Regulatory Commission, "Recommendations for Enhancing Reactor Safety in the 21st Century, the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," Commission Paper SECY-11-0093, July 12, 2011 (ADAMS Accession No. ML11186A950).

U.S. Nuclear Regulatory Commission Letter to All Power Reactor Licensees et al., "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendation 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 12, 2012 (ADAMS Accession No. ML12053A340).

U.S. Nuclear Regulatory Commission, "Nuclear Regulatory Commission (NRC) Comments on NEI 12-13, 'External Hazards PRA Peer Review Process Guidelines'" NRC Staff letter, dated November 16, 2012, ADAMS Accession No. ML12321A280.

APPENDIX A

POSITIONS FROM THE SCREENING, PRIORITIZATION AND IMPLEMENTATION DOCUMENT

APPENDIX A: POSITIONS FROM THE SCREENING, PRIORITIZATION AND IMPLEMENTATION DOCUMENT

This interim staff guidance (ISG) incorporates the guidance from the screening, prioritization, and implementation document (SPID) on six technical elements. The SPID guidance is further discussed below for three technical topics incorporated into this ISG. In addition, guidance on the topics of potentially high frequency sensitive equipment and SSC screening methods and levels is provided in Sections 4.2.5 and 4.5.2, respectively.

The topics covered below are:

- use of existing structural models
- scaling of in-structure response spectra
- use of fixed-based structural models for soft rock

Use of Existing Structural Models

The development of in-structure response spectra (ISRS) is required for both seismic margin assessment (SMA) and seismic probabilistic risk assessment (SPRA). Using existing structural models, where appropriate, will facilitate the timely completion of the SPRA/SMA effort within the desired accuracy required as part of the response to the 50.54(f) letter. Industry and the NRC have agreed that in some cases existing structural models (i.e., those used for design basis or in USI-A-46/IPEEE studies) could be used in structural dynamic analyses performed to support SPRAs or SMAs required as part of the response to the 50.54(f) letter. However, not all models have the appropriate attributes, or are of sufficient complexity, to adequately capture the structural response.

Therefore, an experienced structural engineer(s) (and a peer reviewer) must perform a review of each of the existing models to determine the adequacy of the models for dynamic analysis for application in risk assessments conducted for addressing Recommendation 2.1. If necessary, the existing structural models can be enhanced to bring it to an acceptable level. Industry and the NRC agreed to a set of criteria (provided below) to determine if an existing model can be used directly, or if it must be enhanced or replaced.

Each licensee will need to demonstrate and document that its models are adequate for addressing the 50.54(f) letter and meet the criteria in the SPID (if a new model is not developed using current practice). Any potential structural issues including the adequacy of the model should be addressed and justified in the documentation. The model itself, the modeling process, and the documentation should be subject to peer review, which also will be documented.

The criteria against which structural engineer(s) and peer reviewer(s) should review the existing models are listed below.

(1) The structural models should be capable of capturing the overall structural responses for both the horizontal and vertical components of ground motion.

(2) If there is significant coupling between the horizontal and the vertical responses, one combined structural model should be used for analyzing all three directions of the

earthquake. See ASCE 4-98 Section 3.1.1.1 “Models for Horizontal and Vertical Motions.”

(3) Structural mass (total structural, major components, and appropriate portion of live load) should be lumped so that the total mass, as well as the center of gravity, is preserved. Rotational inertia should be included if it affects the response in the frequency range of interest. See ASCE 4-98 Section 3.1.4.1, “Discretization of Mass” Part (b) 1.

(4) The number of nodal or dynamic degrees of freedom should be sufficient to represent significant structural modes. All modes up to structural natural frequencies of about 20 Hz in all directions should be included (vertical floor slab flexibility generally will not be considered because it is expected to have frequencies above 15 Hz). This will ensure that the seismic responses and ISRS developed in the 1 to 10 Hz frequency range are reasonably accurate. See ASCE 4-98 Section 3.1.4.1, “Discretization of Mass,” Part (b) 2.

(5) Torsional effects resulting from eccentricities between the center of mass and the center of rigidity should be included. The center of mass and the center of rigidity may not be coincident at all levels, and the torsional rigidity should be computed. See ASCE 4-98 Section 3.1.8.1.3, “Requirements for Lumped-mass Stick Models [LMSM],” Parts (b) and (c). Alternatively, a multiple LMSM may be used if the stiffness elements are located at the centers of rigidity of the respective groups of element and the individual models are properly interconnected.

(6) The analyst should determine if one stick model sufficiently represents the structure. For example, two stick models could be appropriate for the analysis of internal and external structures of the containment founded on a common mat.

(7) The structural analyst should review whether in-plane floor flexibility (and subsequent amplified seismic response) has been captured appropriately for developing accurate seismic response up to the 15 Hz frequency. Experience has shown that for nuclear structures with floor diaphragms that have length to width ratios greater than about 1.5, the in-plane diaphragm flexibility may need to be included in the LMSM. As with all these recommendations, alternate approaches can be used when justified.

Scaling of In-Structure Response Spectra

The NRC staff and industry have agreed that scaling approaches can be used in developing ISRS for those cases in which the new site-specific hazard spectral shape is approximately similar to the spectral shape previously used to generate the ISRS. The use of scaling will reduce the effort involved in performing detailed soil structure interaction analyses for the new hazard response spectrum, facilitating the timely completion of the SMA and SPRA efforts for those plants that are screened-in.

Guidance on scaling is provided in industry documents such as EPRI report NP-6041-SL Revision 1 and EPRI report 103959. Scaling of ISRS is an accepted approach that has been used in previous SMA and SPRAs, including the recent Surry pilot SPRA. An example approach for the scaling of “non-similar” shapes conducted for the Surry pilot SPRA project will be described in the SPID. Unfortunately, hard and fast rules as to what is “close enough” are hard to come by. The NRC staff and industry have agreed that it is not possible at this time to provide more than general guidance (with examples of what clearly is and what clearly is not acceptable). The SPID provides examples of pairs of spectra that are and are not sufficiently similar to justify the use of scaling.

The acceptability of scaling of responses will be based on the following:

- previously developed ISRS
- shapes of the previous input response spectrum or review level earthquake (RLE) ground motion
- shapes of the new RLE ground motion, and the structural natural frequencies, mode shapes, and participation factors

Licensees will need to demonstrate and document that ISRS scaling is appropriate for the site and each applicable structure in their submission to the NRC. Any potential structural issues with the use of scaling should be addressed and justified in the documentation. The use of scaling and the documentation should be subject to peer review, which also will be documented.

Scaling of rock or soil sites, where the shape of the new hazard spectrum is not highly similar to the previous spectrum, is not recommended without justification that demonstrates the validity of the scaling approach.

Use of a Fixed-Base Structural Model for Soft Rock Conditions

Some existing structural models and ISRS were developed using an earlier definition of rock ($V_s \geq$ 5,000 ft/sec), which is now considered a soft rock. Based on recent analyses discussed in the SPID, for purposes of the seismic reevaluation process, this earlier definition of rock can be used for the development of the ISRS because past analyses and experience has shown that the amplified response spectra in the 1-10 Hz rock-founded structures fare approximately the same from a fixed based model and a model that uses soil-structure interaction analysis. Therefore, for most rock-founded structures, it is conservative to use fixed base dynamic analyses even when the shear wave velocities are not as high as the current definition of rock (of $V_s \geq$ 9200 ft/sec). The exception may be for structures with high frequency first modes. It is also acceptable for some nuclear power plants located on rock with a $V_s \geq$ 3,500 ft/sec to use a fixed base model, depending on properties of the site and the structure.

For the purpose of addressing the seismic reevaluation process, fixed based models can be used in the dynamic analyses of rock-founded structures with $V_s \geq$ 3,500 ft/sec, provided that justification is provided. Sites founded on rock with V_s between 3,500 and 5,000 ft/sec should pay special attention to the potential issues discussed in the SPID.

Additional Reference

American Society of Civil Engineers 4-98, “Seismic Analysis of Safety-Related Nuclear Structures, American Society of Civil Engineers,” Standard ASCE 4-98, 1998.