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Protecting People and the Environment

<u>JAPAN LESSONS-LEARNED PROJECT DIRECTORATE</u>

JLD-ISG-2012-04

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

DRAFT Interim Staff Guidance
Revision 0
(DRAFT Issue for Public Comment)

JAPAN LESSONS-LEARNED PROJECT DIRECTORATE

JLD-ISG-2012-04

Enhancements to the NRC Method for Seismic Margins Assessments for Response to the March 2012 Request for Information

DRAFT Interim Staff Guidance

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INTERIM STAFF GUIDANCE (ISG)
JAPAN LESSONS-LEARNED PROJECT DIRECTORATE (JLD)

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 that 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 to allow the use of recent advances in both methods and guidance, including guidance in the 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, and the Screening, Prioritization, and Implementation document (SPID) currently under development by industry for NRC endorsement[1].

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 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 the licensing of new reactors. 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 these requirements. 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 on March 11, 2011, the NRC established a senior-level agency task force referred to as the Near-Term Task Force (NTTF). The NTTF was tasked with conducting a systematic and methodical review of NRC regulations and processes and determining if the agency should make additional improvements

[1] The SPID is expected to be finalized before the issuance of the final ISG and the ISG document may be updated if necessary (August 21, 2012 version of the SPID is available at ML12236A362). Public interactions between NRC staff and industry on the development of the SPID are ongoing and will continue up to the issuance of the SPID later this year.

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. These recommendations were enhanced by the NRC staff following interactions with stakeholders. Documentation of the staff's efforts is contained in 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.

As directed by 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 NRC to implement the recommendations. SECY-11-0124 and SECY-11-0137 established the staff's prioritization of the NTTF recommendations.

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

Enclosure 1 to the 50.54(f) Request for Information Letter states that, "The 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 (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 to allow use of the more recent advances in methods and guidance, including guidance in the ASME/ANS standard and the Screening, Prioritization, and Implementation document currently under development by industry for NRC endorsement.

There are currently three methods that could be used to perform an SMA: (1) the PRA-Based SMA method (as described in NRC Interim Staff Guidance (ISG) DC/COL-ISG-020), (2) the NRC SMA method (as described in NUREG/CR-4334, supplemented by NUREG/CR-4482 and NUREG/CR-5076), and (3) the EPRI SMA method (as described in EPRI NP-6041-SL Revision 1 (EPRI, 1991)). These three methods differ in two key areas: the initiators considered in the analysis and the system logic model approach employed.

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 does not achieve the objectives described the 50.54(f) letter due to the use of success paths. In principle, the full PRA-based SMA can be used; however, DC/COL-ISG-020 is structured for use in the licensing of new reactors. In addition, DC/COL-ISG-020 does not address some specific considerations that are pertinent to the 50.54(f) request.

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

Final Resolution

The contents of 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 Take 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.

Draft

Enhancements to the NRC 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, that 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 to allow for the use of recent advances in methods and guidance, including guidance in the 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, (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 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 the licensing of new reactors. 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 these requirements. 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.

<u>Accident Sequence</u> – 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).

<u>Accident Sequence Analysis</u> – The process to determine the combinations 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 CDFM (Conservative Deterministic Failure Margin) methodology described in EPRI Report NP-6041-SL Revision 1 (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 occurring before the effective implementation of off-site 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 certain depth/location) that is 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) is typically 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) Request for Information Letter, the RLE is the envelope of the SSE and the GMRS, as discussed in Section 4.3.1. The RLE should be applied at the location of the GMRS.

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

The "Seismic Margin Assessment" (SMA) method for assessing the capability of nuclear power plants to withstand earthquakes above their design basis was initially developed in 1984-85 by a panel of experts supported by the NRC. The SMA method was originally described in NUREG/CR-4334, which has been further supplemented by NUREG/CR-4482 and NUREG/CR-5076. Shortly after the development of the SMA method by the NRC, EPRI developed a related but different SMA methodology, as described in Electric Power Research Institute (EPRI) Report NP-6041-SL (EPRI, 1991). These two methods have are commonly called the "NRC SMA method" and "EPRI SMA method", respectively.

In order to assess the methods, a trial of the NRC SMA method was conducted at the Maine Yankee Nuclear Power Plant (NUREG/CR-4826), a trial of the EPRI SMA method was conducted at the Catawba Nuclear Station (EPRI Report NP-6359 (EPRI, 1988)), and trials of both methods were conducted concurrently for the Hatch Nuclear Plant (NUREG/CR-5632). After these three trial applications, the NRC endorsed using either the NRC or EPRI methods when the Individual Plant Examination for External Events (IPEEE) program was undertaken by all of the licensed nuclear power plants in response to Supplement 4 to NRC Generic Letter 88-20. However, in the IPEEE staff guidance in NUREG-1407, the NRC required certain enhancements to the SMA methods if either was to be used for responding 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 remainder 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 and the EPRI SMA methodologies is in their approaches to the SMA systems analysis. The systems analysis is used to develop the seismic equipment list (SEL) that is the basis for the seismic fragilities part of the SMA methodology. The NRC SMA method as it was described in NUREG/CR-4334, employs a seismic PRA fault-tree and event-tree approach to delineate accident sequences, albeit limited to only a selected number of safety functions necessary to prevent core-damage. The EPRI SMA method employs a success-path approach in which two success paths are the basis for the systems analysis. This approach defines and evaluates the high confidence of low probability of failure (HCLPF) capacity of those structures, systems, or components (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, such as 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 categories of SSCs. 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"[2] of an individual SSC. The EPRI method uses the CDFM (Conservative Deterministic Failure Margin) method for determining the HCLPF capacity, whereas the NRC method permits the CDFM method but prefers use of the FA (Fragility Analysis) 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 full fragility curve from which the HCLPF capacity is extracted.

It should be noted that Part 5 of the ASME/ANS PRA Standard, endorsed by the NRC in RG 1.200, would permit use of the CDFM method to derive the SPRA fragilities for some applications (see note 1 to supporting requirement FR-F1). The fragility curves 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 required certain enhancements if an SMA analysis was to be used in the IPEEE. Today, based on the experience gained over the two intervening decades, additional enhancements are judged by the NRC staff 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

[2] See Section 2 of this document for a definition of the High Confidence Low Probability of Failure (HCLPF) capacity.

Below is a list of the high level features and enhancements that are necessary if an SMA is to be used for the purposes of responding to the 50.54(f) letter. Some of these topics are similar to staff positions taken during the IPEEE program, and 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 either the original NRC SMA guidance (in NUREG/CR-4334 and NURE/CR-5076) or the NRC's IPEEE guidance (in NUREG-1407) (see Section 4.4 for more detail).
- The scope should include certain containment and containment systems, so as 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.
- Separately report HCLPF capacities for those sequences with non-seismic failures and human actions and HCLPF capacities for those sequences without them.
- The SMA analysis should assume that the earthquake causes an unrecoverable loss of offsite power. (The same assumption was required for the IPEEE, and is a standard assumption in all SMAs).

4.0 <u>Staff Positions on Individual Technical Issues</u>

4.1 Introduction

4.1.1 Organization of the staff positions provided

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 (SPID)

NRC staff is currently engaged with industry in development of guidance for some specific technical elements needed to address the 50.54(f) letter. This applicable guidance is to be provided in a document called the Screening, Prioritization, and Implementation Document (SPID), which has an expected publication date of November 2012. The SPID is being developed by industry with a significant level of review and input by NRC staff; and it is anticipated that the SPID will be submitted to the NRC by the Nuclear Energy Institute, and that NRC will endorse the SPID after NRC staff review. To the extent appropriate, the applicable draft SPID positions are incorporated into this draft ISG document, and some additional information is also provided in Appendix A. *[The SPID positions are expected to be finalized before the issuance of the final ISG and the ISG document may be updated if necessary (August 21, 2012 version available at ML12236A362)]*

4.2 Seismic Margin Assessment Scope Issues

4.2.1 Introduction

There are currently three methods that could be used to perform an SMA: the PRA-Based SMA method (as described in NRC Interim Staff Guidance (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 (EPRI, 1991)). The three methods differ in two key areas: the initiators considered in the analysis and the system logic model approach employed. 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/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 due to its use of a success path approach (as discussed above). In principle, the full PRA-based SMA can be used; however, DC/COL-ISG-020 is structured for use in the licensing of new reactors. In addition, DC/COL-ISG-020 does not address some specific considerations that are pertinent to the 50.54(f) request.

In the IPEEE program, a tiered approach was taken and three “scope” levels were developed for use by plants taking into account estimates of site-specific hazard levels. The definitions of for these levels, termed “reduced”, “focused” 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) program implies use of the NRC SMA with the enhancements described in this document.

The applicability of the SMA method for the purposes of addressing the 50.54(f) letter for each site will be determined by NRC staff during the screening and prioritization phase of the 50.54(f) Recommendation 2.1 program. 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 that are 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. For an SMA, this means that the analyst is required to determine and report the HCLPF capacities of these two categories of sequences separately.
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 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." One way to include non-seismic factors in the quantification is to use the convolution method (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. Previous guidance for the IPEEE program was provided in NUREG 1407 for full scope plants. Guidance is also provided in the AMSE-ANS PRA standard in Part 5 and Part 10. 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 2 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 SSE 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 will also provide additional guidance and component capacity information useful to the plants that are 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 will also 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 related guidance documents are endorsed by the NRC, 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 employs a Review Level Earthquake (RLE)[3] 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/location) that is used as the basis for the analyses performed in a Seismic Margin Assessment.

While the same term was used in the IPEEE program, the method to be used to determine the RLE for the purposes of responding to the 50.54(f) letter is different. The 50.54(f) letter specifies that the licensee compare the site-specific Ground Motion Response Spectrum (GMRS)[4] with the plant's Safe Shutdown Earthquake (SSE) ground motion response spectrum and use the envelope of these two spectra as the RLE. The RLE should be applied at the location of the GMRS. 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.

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 a fragility 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.

[3] In some documents, the term "Seismic Margin Earthquake" or SME may be used.
[4] Acceptable methods for development of the GMRS are described in the 50.54(f) letter.

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. An 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 CDF, LERF, 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, the IPEEE, or the 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 assure that the objectives of the 50.54(f) letter are achieved.

In an effort to appropriately optimize the use of existing information, NRC and industry experts have been working to develop guidance on several aspects of structural response that will be included in the SPID. This draft guidance is discussed in detail in Appendix A. 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 for the 50.54(f) program if they have sufficient complexity and attributes to provide the appropriate level of accuracy needed for the SMA (or PRA). Specific criteria are provided in Appendix A. 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 below is followed.

The use of any existing models or data should be reviewed by an experienced structural engineer, and should be subject to peer review. Use of existing information should be documented in the submission to the NRC and should be adequately justified. Any potential issues or deficiencies should be addressed in the documentation and justification provided.

Staff Position: Realistic ISRS should be calculated using ASME/ANS PRA Standard Part 10 or the guidance on the use of existing information and models provided in the SPID (See Appendix A). If an existing structural model is used, its attributes should be compared to the criteria in the SPID and its applicability documented and justified. If scaling of ISRS is used, its use should be consistent with current accepted practice or the SPID guidance on the use of scaling. The use of scaling should be documented and justified. Fixed base models may be used for structures founded on rock with a shear wave velocity greater than 3,500 feet/second.

The use of any existing models or data should be reviewed by an experienced structural engineer, and should be subject to peer review. Use of existing information should be documented in the submission to the NRC and should be adequately justified. Any potential issues or deficiencies should be addressed in the documentation and justification provided.

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 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 of 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. Certain enhancements were provided by the NRC staff in the IPEEE guidance (NUREG-1407). For an SMA performed to address the 50.54(f) letter, the scope must be extended in order to identify sequences involving a potential large early radioactivity release. To accomplish this expanded scope, certain features and enhancements are considered necessary. 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 PWRs and NUREG/CR-5076 for 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 in order 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.

Successes: The systems model should retain as "successes" those SSCs screened out as having strong seismic capacity in order to facilitate review of the accident sequences. For these SSCs, it is acceptable to use an estimated or generic HCLPF capacity. Further guidance is provided in Section 4.5.2.

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 achieve understanding of 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. Note that the scope of this type of analysis is

greater than that for an SMA performed under the IPEEE guidance (NUREG-107), 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 systems-analysis part of an SMA involves, in part, the selection of a seismic equipment list (SEL). This is a standard aspect of both SMA and seismic PRA. For purposes of 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 4.4.1 above. The starting point for constructing the SEL is the internal-events PRA model, to which must be added a number of SSCs with earthquake-specific issues, such as including 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 purposes of the walkdown are to ensure that the seismic fragilities or margins are realistic and plant-specific, and to find any as-designed, as-built, and as-operated seismic vulnerabilities in the plant. It should be done in sufficient detail and documented in a sufficiently complete fashion so that the subsequent screening or margin evaluation is traceable. For guidance on walkdown, the licensee is referred to EPRI NP-6041 and NUREG/CR-4334.

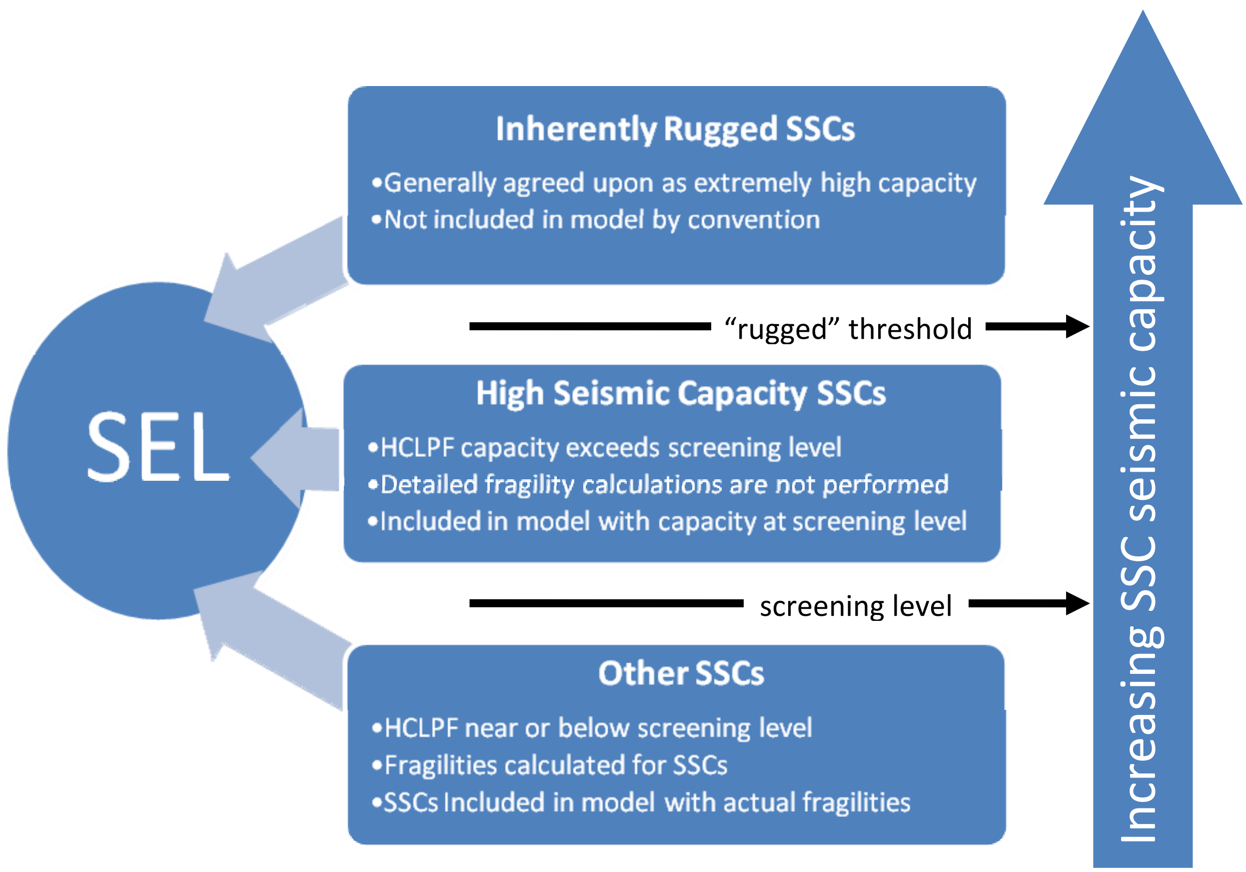
A set of technical requirements on the walkdown are found in EPRI NP-6041 and in Part 10 of ASME/ANS PRA Standard, which refers back to EPRI NP-6041.

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

4.5.2 Screening approach and level for of SSCs

Technical Issue:

The SSCs in the SEL can generally be broken into three categories or "bins" based on their seismic capacity as it compares 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 in terms of both the approach used to determine their capacity and in terms of 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 that are believed to have a seismic capacity beyond any realistic earthquake loading levels (e.g., manual valves). These components are generally mutually agreed upon by the technical community and, by convention, they are not included in SMA or SPRA models. Guidance on inherently rugged components can be found in EPRI NP-6041-SL.

Other SSCs may be less rugged but still have sufficient capacity such that their failures would be unlikely to contribute significantly to the SCDF 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 TR-103959 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 may also 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 that are 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, where 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 SSCs that were the most risk significant.

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

In the 50.54(f) program, the risk information needed includes CDF, LERF, and the identification of the dominant risk contributors, as indicated in the March 12, 2012 letter. In order to be able to gain the necessary risk insights from an SMA, the screening level must be set well above the RLE. Based on analysis 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 a 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 determine and judge if explicit fragility/HCLPF calculations are needed for each SSC.

SSCs identified as high-seismic capacity components must be included in the model and the capacity should be set at the screening level.

The screening tables of NUREG/CR-4334 and EPRI-NP-6041 can be used for identifying and assigning conservative HCLPF values to the high seismic capacity SSCs using a screening level that is higher than the RLE as defined above. Use of these tables must include satisfying caveats associated with the tables and should include anchorage evaluations, as appropriate. This is an enhancement from earlier guidance, and from 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 assure that none of the components that are 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 assure that each of the non-seismic failures that are 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. Guidance given in EPRI NP-6041-SL Rev.1 and NUREG/CR-4334 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 high capacity SSCs should be assigned capacities equal to the screening level. 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 a 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 assure that none of the following conditions exist:
  - A high seismic capacity SSC is identified as a dominant contributor to HCLPF of CDF.
  - A high seismic capacity SSC 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 rerun.

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

Technical Issue

The Fragility Analysis method is described in a number of references, for example, NUREG/CR-2300, 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

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

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% conditional-probability-of-failure value i.e.,

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

For every component in the plant system model, the fragility analysis 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.

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.

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

Technical Issue The fragility analysis method of estimating the HCLPF capacity of SSCs, although universally applicable, does however require the median factors of safety for different variables affecting the response and capacity 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 a deterministic procedure called the “Conservative Deterministic Failure Margin (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 in 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% exceedance actual strength if test data are available; 84% non-exceedance ground response spectrum etc.). The calculation of HCLPF 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: In the convolution method, accident sequences are evaluated by combining input fragility curves according to the Boolean expression for each sequence. Seismic and random/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% with a confidence level of 95%. This way of developing the plant level fragility curves will retain the information for the analyst and NRC to develop accident sequence frequencies, CDF and LERF calculations using the site-specific seismic hazard as needed. The procedure 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 will also 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%.

In order 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. The median capacity is calculated using the equation

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

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: 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, β_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% confidence of not more than 5% 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, section 2) is the way that an SMA analysis derives an approximate sequence-level HCLPF capacity from the HCLPF capacities of individual SSCs that comprise the accident sequence, or derives an approximate plant-level HCLPF capacity from the most important (lowest) sequence-level HCLPF capacities that emerge from the SMA analysis.

However, the "min-max" method is sometimes only a rough approximation and it gives a 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 distortion is not important, however, if a single SSC "dominates" the HCLPF capacity of an accident sequence, in the sense that no other SSC HCLPF capacities are "close," or if a single accident sequence "dominates" the overall seismic risk profile.

The existence of this potential distortion requires the SMA analyst to be alert to the problem. In cases like those mentioned, it is not difficult for the SMA analyst to perform the correct "arithmetic" to derive a sequence-level HCLPF capacity that is very much closer to the true value, or to derive a much less approximate plant-level HCLPF capacity.

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

5.0 Staff Positions on Documentation

5.1 Documentation Content

Technical Issue: In order to address the request in the 50.54(f) letter, the SMA must be appropriately conducted and documented. Documentation to be submitted to the NRC includes a number of elements, as shown below.

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

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

(2) The definition of the response spectrum shape used for the fragility analysis 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 that were 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 both for 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 reports, and
- the results of the evaluation.

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 manner 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[5]:

Peer review should include the following attributes:

- The peer review should be a participatory peer review, rather than a late-stage review.
- 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 development of the SEL), 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 where 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 SQUG Walkdown Screening and Seismic

[5] The above staff positions, which may differ from the endorsed peer review process set forth in the ASME/ANS PRA Standard, are applicable to SMAs performed pursuant to these 50.54(f) letters and does not represent a change in any regulatory positions set forth in RG 1.200.

Evaluation Training Course or equivalent, or shall have demonstrated experience in seismic walkdowns.

- 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 should also 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.
- Information as to the disposition of comments.
- A description of the peer review process.
- A discussion of the key findings and a discussion as to how the findings were addressed.
- The peer review should include 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.

Electric Power Research Institute, “Seismic Probabilistic Risk Assessment Implementation Guide,” EPRI report 1002989, Palo Alto, California, 2003.

Electric Power Research Institute, “Seismic Fragility Application Guide,” EPRI 1019200, Palo Alto, California, 2009.

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, 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, 2009, ADAMS Accession No. ML090410014.

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 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.

APPENDIX A: DRAFT POSITIONS FOR THE SCREENING, PRIORITIZATION AND IMPLEMENTATION DOCUMENT

This ISG incorporates the draft guidance from the SPID on 6 technical elements. The draft guidance for the SPID is discussed below for 3 technical topics of interest to 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 SMA and SPRA. Using the 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 that are 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, a review of each of the existing models must be performed by an experienced structural engineer(s) (and a peer reviewer) 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 whether or not an existing model can be used directly, or whether it must be enhanced or replaced. *[At the time of this writing, detailed guidance to further describe the criteria below is being developed for the SPID. This guidance is expected to be finalized in October 2012.]*

Each licensee will need to demonstrate/document that their 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 will also 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 will generally 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” 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 the purposes of 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

NRC staff and industry have agreed that scaling approaches can be used in developing ISRS for those cases where 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 (SSI) 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 Rev. 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. 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:

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

Licensee will need to demonstrate/document that scaling of the ISRS 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 will also 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 the original definition of rock ($V_s \geq$ 3,500 ft/sec), which is now considered a soft rock. For purposes of the 50.54(f) program, 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 (SSI) analysis. Therefore, for the majority of 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.

For the purpose of addressing the 50.54(f) program, fixed based models can be used in the dynamic analyses of rock-founded structures using this earlier definition of rock.

Additional References

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