UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

Month XX, 2011

NRC GENERIC LETTER 2011-XX: SEISMIC RISK EVALUATIONS FOR OPERATING REACTORS

ADDRESSEES

All holders of an operating license or construction permit for a nuclear power reactor issued under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," except those who have permanently ceased operation and have certified that fuel has been removed from the reactor vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter (GL) for the following purposes:

- to request that addressees evaluate their facilities to determine the current level of seismic risk

- to collect the requested information to facilitate NRC's determination if there is a need for additional regulatory action

Pursuant to 10 CFR 50.54(f), addressees are required to submit a written response to this GL.

BACKGROUND

Structures, systems, and components (SSCs) important to safety at nuclear power reactors must be designed to withstand the effects of natural phenomena, including earthquakes, without losing the capability to perform their intended safety functions. SSCs in operating nuclear power plants are designed either in accordance with, or have been revised to meet the intent of Appendix A to 10 CFR Part 100 and Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 2. The state of knowledge of seismic hazard within the United States has evolved to the point that the NRC has concluded that, in view of the potential safety significance of this issue, it is necessary to reexamine the level of conservatism in the determination of original seismic design estimates. Analyses performed under the Generic Issue program (GIP) indicated the need to evaluate in more detail the impact of updated seismic hazard information with respect to operating commercial nuclear reactors. The background information relevant to this GL includes the individual plant examinations of external events (IPEEEs) and Generic Issue (GI)-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," dated June 9, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML051600272). The following paragraphs summarize these two studies.

Individual Plant Examination of External Events

On June 28, 1991, the NRC issued Supplement 4 to GL 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," (ADAMS Accession No. ML031150485) to request that each licensee identify and report to the NRC all plant-specific vulnerabilities to severe accidents caused by external events. The IPEEE program included the following four supporting objectives:

(1) Develop an appreciation of severe accident behavior.

(2) Understand the most likely severe accident sequences that could occur at the licensee's plant under full-power operating conditions.

(3) Gain a qualitative understanding of the overall likelihood of core damage and fission product releases.

(4) Reduce, if necessary, the overall likelihood of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

The external events to be considered in the IPEEE were seismic events; internal fires; and high winds, floods, and other external initiating events, including accidents related to transportation or nearby facilities and plant-unique hazards.

In June 1991, at about the same time the NRC issued Supplement 4 to GL 88-20, the NRC issued NUREG-1407, "Procedure and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," (ADAMS Accession No. ML063550238) which provided guidelines for conducting IPEEEs. On September 8, 1995, the NRC issued Supplement 5 to GL 88-20 (ADAMS Accession No. ML031130465) to notify licensees of modifications to the recommended scope of the seismic portion of the IPEEE for certain plant sites in the Central and Eastern United States (CEUS).

NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," issued April 2002, (ADAMS Accession Nos. ML021270070 and ML021270674) provides insights gained by the NRC from the IPEEE program. Almost all licensees reported in their IPEEE submittals that no plant vulnerabilities were identified with respect to seismic risk (the use of the term "vulnerability" varied widely among the IPEEE submittals). However, most licensees did report at least some seismic "anomalies," "outliers," or other concerns. In the few submittals that did identify a seismic vulnerability, the findings were comparable to those identified as outliers or anomalies in other IPEEE submittals. Seventy percent of the plants proposed improvements as a result of their seismic IPEEE analyses. In several responses, neither the IPEEE analyses nor subsequent assessments documented the potential safety impacts of these improvements, and in most cases, plants have not reported completion of these improvements to the NRC.

Generic Issue 199

In support of early site permits (ESPs) and combined license applications (COLs) for new reactors, the NRC staff reviewed updates to the seismic source and ground motion models provided by applicants. These seismic updates included new Electric Power Research Institute models to estimate earthquake ground motion and updated models for earthquake sources in

the CEUS, such as around Charleston, SC, and New Madrid, MO. These reviews identified higher seismic hazard estimates than previously assumed that may result in the increased likelihood of exceeding the safe-shutdown earthquake (SSE) at operating facilities in the CEUS. The staff determined that based on the evaluations of the IPEEE program, seismic designs of operating plants in the CEUS do not pose an imminent safety concern. At the same time, the staff also recognized that, because the probability of exceeding the SSE at some currently operating sites in the CEUS is higher than previously understood, further study was warranted. As a result, the staff concluded on May 26, 2005 (ADAMS Accession No. ML051450456), that the issue of increased seismic hazard estimates in the CEUS be examined under the GIP.

GI-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants" was established on June 9, 2005 (ADAMS Accession No. ML051600272). The initial screening analysis for GI-199 suggested that estimates of the seismic hazard for some currently operating plants in the CEUS have increased. The NRC completed the initial screening analysis of GI-199 on February 1, 2008 (ADAMS Accession No. ML073400477), which concluded that GI-199 should proceed to the safety/risk assessment stage of the GIP. The NRC held a public meeting on February 6, 2008 (ADAMS Accession No. ML080350189), at which the NRC staff discussed its ongoing activities related to GI-199, described the screening process and criteria, and explained the screening analysis results.

Subsequently, during the safety/risk assessment stage of the GIP, the NRC staff reviewed and evaluated the new information received with the ESP/COL submittals, along with 2008 U.S. Geological Survey seismic hazard estimates and recent geological research literature. The staff compared the new seismic hazard data with the earlier evaluations conducted as part of the IPEEE program. From this evaluation, the staff concluded that the likelihood of exceeding the seismic hazard used in the IPEEE program could be higher than previously understood for some currently operating CEUS sites.

The NRC staff completed the safety/risk assessment stage of GI-199 on September 2, 2010 (ADAMS Accession No. ML100270582), concluding that GI-199 should transition to the regulatory assessment stage of the GIP. The NRC staff presented this conclusion at a public meeting held on October 6, 2010 (ADAMS Accession No. ML102950263). Information Notice (IN) 2010-018, "Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,'" dated September 2, 2010 (ADAMS Accession No. ML101970221) summarizes the results of the GI-199 safety/risk assessment.

APPLICABLE REGULATORY REQUIREMENTS

- Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, GDC 2, "Design Bases for Protection against Natural Phenomena"

- 10 CFR 50.54, "Conditions of Licenses"

- 10 CFR 50.109, "Backfitting"

- 10 CFR 50.34(a)(1), (a)(3), (a)(4), (b)(1), (b)(2), and (b)(4)

- Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria"

The seismic design bases for currently operating nuclear power plants were either developed in accordance with, or have been revised to meet the intent of GDC 2 and 10 CFR Part 100, Appendix A. GDC 2 requires the design basis for SSCs to reflect appropriate consideration of the most severe of the natural phenomenon that have been historically reported for the site and surrounding area, with sufficient margin. Although the regulatory requirements in Appendix A to 10 CFR Part 100 are fundamentally deterministic, the current NRC process for determining the seismic design bases at new reactor sites is fundamentally probabilistic. The seismic risk evaluation process is inherently probabilistic in nature.

On August 16, 1995, the NRC published a final policy statement entitled, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (Volume 60, page 42622, of the *Federal Register* (60 FR 42622)), which encouraged the use of probabilistic risk assessment (PRA) methods and stated that "PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109, 'Backfitting.'" NRC regulations and guidance such as 10 CFR 50.109 and NUREG/BR-0058 Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," (ADAMS Accession No. ML042820192) provide a framework for changing regulatory positions in light of new information. The GIP utilizes the "Backfitting" and regulatory analysis guidelines for evaluating generic issues.

DISCUSSION

GI-199 was initiated because of the need to evaluate the effect of updated seismic hazard estimates on operating nuclear power plants. The GI-199 safety/risk assessment investigated the safety and risk implications of updated earthquake-related data and models. These data and models suggest that the probability for earthquake ground shaking above the seismic design basis for some nuclear power plants in the CEUS is greater than previous estimates.

In the safety/risk assessment, the NRC staff used the risk metric of the change in seismic core damage frequency (SCDF) derived from an updated understanding of the site-specific seismic hazard estimates from those previously used in the IPEEE submittals. The changes in SCDF estimate in the safety/risk assessment for some plants lie in the range of 10^{-4} per year to 10^{-5} per year, which meet the numerical risk criteria for an issue to continue to the regulatory assessment stage of the GIP.

It is recognized that the approach used to estimate SCDF in the safety/risk assessment was not based on a rigorous methodology. The approach merely extrapolated from the information available within the IPEEE submittals. As described in NUREG-1742, there are limitations associated with utilizing the inherently qualitative insights from the IPEEE submittals in a quantitative assessment. Specifically, the staff's assessment did not provide insight into which SSCs are important to seismic risk. Such knowledge is necessary for the NRC staff to determine, in light of the new understanding of seismic hazards, the safety significance associated with the new information regarding seismic margin. The burden to be imposed by this GL is justified in view of the potential safety significance of this issue.

REQUESTED ACTIONS

The NRC requests that each addressee provide information about the current risk posed by seismic events. Enclosure 1 provides further information on a proposed process, based on a progressive screening approach, to develop the requested seismic risk information. Depending

on the comparison between the updated seismic hazard information and the site-specific SSE, the process is based on a progressive screening approach that allows the use of (1) a Level 1 seismic probabilistic risk assessment (SPRA); or (2) a seismic margin assessment that uses a fault-space based logic model to delineate a broad range of seismic accident sequences (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)), which includes both seismic and non-seismic failures. The review-level earthquake for the seismic margin assessment (SMA) must be a composite spectrum, as described in Enclosure 1.

Consistent with the NRC’s PRA policy statement, the technical adequacy of the methods used to develop the requested information must be sufficient to provide confidence in the results, such that the seismic risk information can be used in regulatory decision-making. Regulatory Guide 1.200 Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” (ADAMS Accession No. ML090410014) provides one acceptable approach for determining the technical adequacy of an SPRA. Part 10 of the American Society of Mechanical Engineers/American Nuclear Society standard, RA-Sa-2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” provides one acceptable approach for determining the technical adequacy of an SMA.

REQUESTED INFORMATION

The NRC requests that each addressee provide the following information (see Enclosure 1 for additional details):

(1) a list of plant seismic vulnerabilities (including any seismic anomalies, outliers, or other findings) identified by the IPEEE and a description of the actions taken to eliminate or reduce them (including their completion dates)

(2) site-specific, base rock, and control point elevation hazard curves (common fractiles and mean) over a range of spectral frequencies (1, 2.5, 5, 10, and 25 Hz and peak ground acceleration - PGA) and annual exceedance frequencies (1×10^{-7} and higher) in tabular and graphical format

(3) the site-specific, performance-based ground motion response spectrum (GMRS) developed from the new site-specific seismic hazard curves in tabular and graphical format

(4) the SSE in tabular and graphical format

(5) The significant contributors to seismic risk, if the GMRS exceeds the SSE. Plants may opt to use either a fault-space based SMA or an SPRA method to evaluate the increased seismic risk.

   A. For plants opting to use the fault-space based SMA, the following information is requested:

      (1) a description of the methodologies used to quantify the seismic margins of high confidence of low probability of failure (HCLPF) capabilities of SSCs, together with key assumptions

(2) a detailed list of the SSC seismic margin values with reference to the method of seismic qualification, the dominant failure modes, the source of information, and the location of each SSC

(3) 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 (C_{50}) and the logarithmic standard deviation or “beta” values) and the technical bases for them

(4) the bases for screening of any SSCs on the safe-shutdown equipment list based on their generic high seismic capacities

(5) a description of the SMA, including the development of its logic models, the safe-shutdown equipment list, the seismic response analysis, the results of the screening analysis, the results of the plant seismic walkdown, the identification of critical failure modes for each SSC, and the calculation of HCLPF capacities for each SSC included in the SMA logic model

(6) a description of the process used to ensure that the SMA is technically adequate, including the dates and findings of peer reviews

B. For plants that opt to perform or update SPRA assessment, the following information is requested:

(1) a list of the significant contributors to SCDF for each seismic acceleration bin, including importance measures (i.e., Risk Achievement Worth, Fussell-Vesely and Birnbaum)

(2) a summary of the methodologies used to estimate the SCDF, including the following:

i. the methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions

ii. SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), the source of information, and the location of the component

iii. the seismic fragility parameters

iv. important findings from plant walkdowns and any corrective actions taken

v. the process used in the seismic plant response analysis and quantification, including the specific adaptations made in the internal events PRA model to produce the seismic PRA model and their motivation

(3) a description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews

REQUIRED RESPONSE

In accordance with 10 CFR 50.54(f), an addressee must respond as described below:

- Within 90 days of the date of this GL, each addressee is requested to submit a written response consistent with the requested information from item 1 above.

- Within 180 days of the date of this GL, each addressee is requested to submit a written response consistent with the requested information from items 2, 3, and 4 above. In its response, where applicable, each addressee is requested to identify its selected assessment approach (i.e., SMA or SPRA).

- Within 1 year of the date of this GL, each addressee that elects to perform an SMA is requested to submit a written response consistent with the requested information from item (5)A above.

- Within 2 years of the date of this GL, each addressee that elects to perform an SPRA is requested to submit a written response consistent with the requested information from item (5)B above.

If an addressee cannot meet the requested response date, the addressee must provide a response within 90 days of the date of this GL and describe the alternative course of action that it proposes to take, including the basis of the acceptability of the proposed alternative course of action and estimated completion dates.

The required written response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, 11555 Rockville Pike, Rockville, MD 20852, under oath or affirmation under the provisions of Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, addressees should submit a copy of the response to the appropriate Regional Administrator.

JUSTIFICATION FOR INFORMATION REQUEST

A significant amount of important research into seismic hazards has been developed over the past 25 years. An assessment of new information indicates a potential for a reduction in safety margin for some currently operating plants in the CEUS, based on a comparison of information provided during the IPEEE effort. The NRC staff developed SCDF estimates for affected plants using new seismic hazard information along with data from the IPEEE submittals. Some of these new SCDF estimates are higher by from 10^{-4} per year to 10^{-5} per year compared to estimates using older seismic hazard information. The NRC does not have sufficient information to determine and quantify, on a plant-specific basis, the safety significance associated with the new information regarding seismic margin.

The NRC is authorized under Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f) to require the addressees of this GL to submit to the NRC the information described in “Requested Response.” The NRC staff has determined that the information collection and reporting burden to be imposed on nuclear power plant licensees by this GL is justified in view of the potential safety significance of the issue of increased seismic hazard estimates and uncertainty in fragility of safety-related SSCs at the addressees’ nuclear power plants.

RELATED GENERIC COMMUNICATIONS

Document Number	Document Name	ADAMS Accession No.
IN 2010-18	Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants"	ML101970221
GL 88-20, Supp 4	Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities	ML031150485

BACKFIT DISCUSSION

This GL contains only the information request described in "Requested Response." The GL does not contain any recommended changes to the design or procedures necessary to operate the nuclear power plants of the addressees. This GL also does not contain any direction or suggestion that the addressees should consider developing or implementing changes to the design or procedures necessary to operate their nuclear power plants in light of the information requested by this GL. The NRC staff does not intend that the probabilistic seismic hazard estimates or the methods of evaluation required by this GL be automatically incorporated into the licensing basis (including design basis) of any of the addressees' nuclear power plants via this GL. The NRC staff is not requiring or recommending the submission of any addressee-initiated changes to the licensing bases for the addressees' nuclear power plants, as the need for such changes will have to be made on a case by case basis by licensees after evaluating the significance of the information developed as a result of this GL.

The NRC will evaluate the information submitted by the addressees in response to this GL and may then determine whether there is a need to take additional action. If that determination results in an action that constitutes an NRC staff recommendation (including the issuance of NRC communications characterized as "guidance") or an NRC requirement (via regulation or order, including licensing action) that one or more of the addressees change the design or the procedures necessary to operate the addressees' nuclear power plants, then the NRC will treat that action as backfitting under the Backfit Rule at 10 CFR 50.109.

Under the provisions of Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended, this GL requests a review and appropriate resulting actions to ascertain whether backfits are warranted. No mandated backfit is intended by the issuance of this GL. Therefore, the NRC staff has not performed a backfit analysis.

***FEDERAL REGISTER* NOTIFICATION**

A notice of opportunity for public comment on this GL was published in the *Federal Register* (76 FR 54507) on September 1, 2011. In addition, the NRC did hold a public meeting on this GL on **MMMM, DD, YYYY (ADAMS Accession No. MLXXXXXXXXX)**.

CONGRESSIONAL REVIEW ACT

This section is not applicable because this proposed GL is being issued for public comment.

PAPERWORK REDUCTION ACT STATEMENT

This GL does not contain new or amended information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing requirements were approved by the Office of Management and Budget, approval numbers 3150-0011 and 3150-0093.

The burden to the public for this mandatory information is estimated to be 1,240 hours per response for plants in the CEUS where the GMRS does not exceed the SSE. Western plants may require an additional 2,500 hours to develop seismic source characterization and ground motion models. For any plant where the GMRS exceeds the SSE, the burden is estimated to be an additional 2,880 hours if the licensee elects to perform an SMA or an additional 3,380 hours if the licensee elects to perform an SPRA. This includes time for reviewing existing data sources, gathering and analyzing the data needed, and completing and reviewing the information collection.

Send comments on any aspect of this information collection, including suggestions for reducing the burden, to the Records and FOIA/Privacy Services Branch (T5-F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001 or by e-mail to infocollects@nrc.gov and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget (OMB), Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

CONTACT

Please direct any questions about this matter to the technical contact or the lead project manager listed below or to the appropriate Office of Nuclear Reactor Regulation project manager.

Timothy J. McGinty, Director
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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under NRC Library/Document Collections.

CONTACT

Please direct any questions about this matter to the technical contact or the Lead Project Manager listed below, or to the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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Development of Requested Information

Introduction

This enclosure describes the suggested process for developing the information requested by the U.S. Nuclear Regulatory Commission (NRC) in Generic Letter (GL) 2011-XX, “Seismic Risk Evaluations for Operating Reactors.” Figure 1 illustrates the process, which is based on an approach that allows the use of either seismic probabilistic risk assessment (SPRA) or seismic margin assessment (SMA) that uses a fault-space systems analysis logic model to delineate a broad range of seismic accident sequences. Addressees may use alternative approaches, if adequately justified in their response to this GL.

The suggested approach shown in Figure 1 is intended to provide information which will allow the NRC to determine the safety significance associated with the new information regarding seismic margin. The approach incorporates a screening step, which compares the safe-shutdown earthquake (SSE) with the ground motion response spectrum (GMRS).

The following paragraphs provide additional discussion about each individual step in Figure 1:

Step 1. NUREG-1742, “Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program,” issued April 2002, indicates that 70 percent of the plants proposed improvements as a result of their seismic individual plant examination of external events (IPEEE) analyses. Implementation of these improvements and any additional subsequent changes that could affect the findings provided during the IPEEE effort must be revisited and identified. This step will achieve a common understanding with addressees about the status of seismic plant improvements made as a result of the IPEEE program.

Step 2. Addressees shall develop site-specific base rock and control point elevation hazard curves (common fractiles and mean) over a range of spectral frequencies (1 Hz, 2.5 Hz, 5 Hz, 10 Hz, and 25 Hz and peak ground acceleration - PGA) determined from a probabilistic seismic hazard analysis (PSHA) as follows:

- Unless a more comprehensive approach can be justified, licensees of plants located in the CEUS are expected to use the Central and Eastern United States Seismic Source Characterization (CEUS-SSC) model and the appropriate Electric Power Research Institute (2004, 2006) ground motion prediction equations. Regional and local refinements of the CEUS-SSC are not necessary for this evaluation.

- Licensees of plants located in the Western United States (Columbia, Diablo Canyon, Palo Verde, and San Onofre) shall develop an updated, site-specific PSHA. Any new or updated seismic hazard assessment should consider all relevant data, models, and methods in the evaluation of seismic sources and ground motion models. The development of the PSHA should follow a structured process consistent with the guidance contained in American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” and NUREG/CR-6372, “Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts.”

For all plants, cumulative absolute velocity filtering is acceptable for scenario earthquakes with moment magnitudes below 5.5. For the site response analysis, site amplification curves should

be developed incorporating appropriate uncertainties. Site response methods 2 or 3, as described in NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines," are appropriate. Site amplification functions should be developed over a broad range of annual frequencies (amplitudes) to facilitate estimation of SCDF. Low- and high-frequency controlling earthquakes at frequencies of 10^{-4} and 10^{-5} per year should be documented.

Addressees shall use the site-specific hazard curves to develop a performance-based GMRS for the site, using the guidance in Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion." The site-specific GMRS shall be determined and clearly specified at the same elevation (clearly specified) as the design-basis SSE assuming a site profile with a free surface above the control point elevation.

Step 3. If the SSE bounds the GMRS at all of the requested frequencies except 25 Hz, then the evaluation may be terminated after the results have been documented (Step 4).

Step 4. Addressees are requested to provide the new seismic hazard curves, the GMRS, and the SSE in graphical and tabular format. For plants with a high frequency ground motion exceedance (25 Hz), the documentation shall also include a detailed justification to confirm that affected plant structures and equipment at various elevations will maintain their seismic function at the higher acceleration levels.

Step 5. This step entails termination of the seismic risk evaluation after demonstrating that sufficient safety margins exist.

Step 6. Addressees can elect to perform a Level 1 SPRA or a fault-space based SMA. Addressees that performed an SPRA as part of their IPEEEs may wish to update it rather than initiating a new type of analysis.

Step 7. Addressees that elect to perform an SPRA shall provide the new seismic hazard curves, the GMRS, and the SSE in graphical and tabular format. In addition, they are requested to provide a plan, including schedule, for completing the SPRA.

Step 8. Addressees that elect to perform an SPRA shall ensure that the SPRA is technically adequate for regulatory decision making. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," provides an acceptable approach for determining the technical adequacy of an SPRA.

Step 9. The results of the SPRA shall be submitted to the NRC for review. The "Requested Information" section in the main body of the GL identifies what specific information should be provided.

Step 10. If SCDF is less than 10^{-5} per year, then NRC may conclude that no further regulatory action is warranted.

Step 11. If SCDF is greater than 10^{-5} per year or the plant level high confidence in low probability of failure (HCLPF) value is less than the review-level earthquake (RLE), addressees are requested to submit plans for actions that lower seismic risk. These actions may include an additional detailed analysis, such as an SPRA, or plant modifications. If the HCLPF is greater than or equal to the RLE, then NRC may conclude that no further regulatory action is warranted.

Step 12. The NRC staff will review the results of the SPRA to identify whether additional regulatory action is warranted.

Step 13. Addressees that elect to perform a fault-space based SMA are requested to provide the new seismic hazard curves, the GMRS, and the SSE in graphical and tabular format. In addition, they are requested to provide a plan, including schedule, for completing the SMA.

Step 14. Addressees that elect to perform a fault-space based SMA must use a composite spectrum RLE, defined as the maximum of the GMRS and SSE at each spectral frequency. ASME/ANS RA-Sa-2009 provides an acceptable approach for determining the technical adequacy of an SMA.

Step 15. The results of the SMA must be submitted to the NRC for review. The "Requested Information" section in the main body of the GL identifies the specific information that should be provided.

Step 16. If the HCLPF values of the structures, systems, and components of a nuclear power plant included in the SMA exceed the RLE, then NRC may conclude that no further regulatory action is warranted.

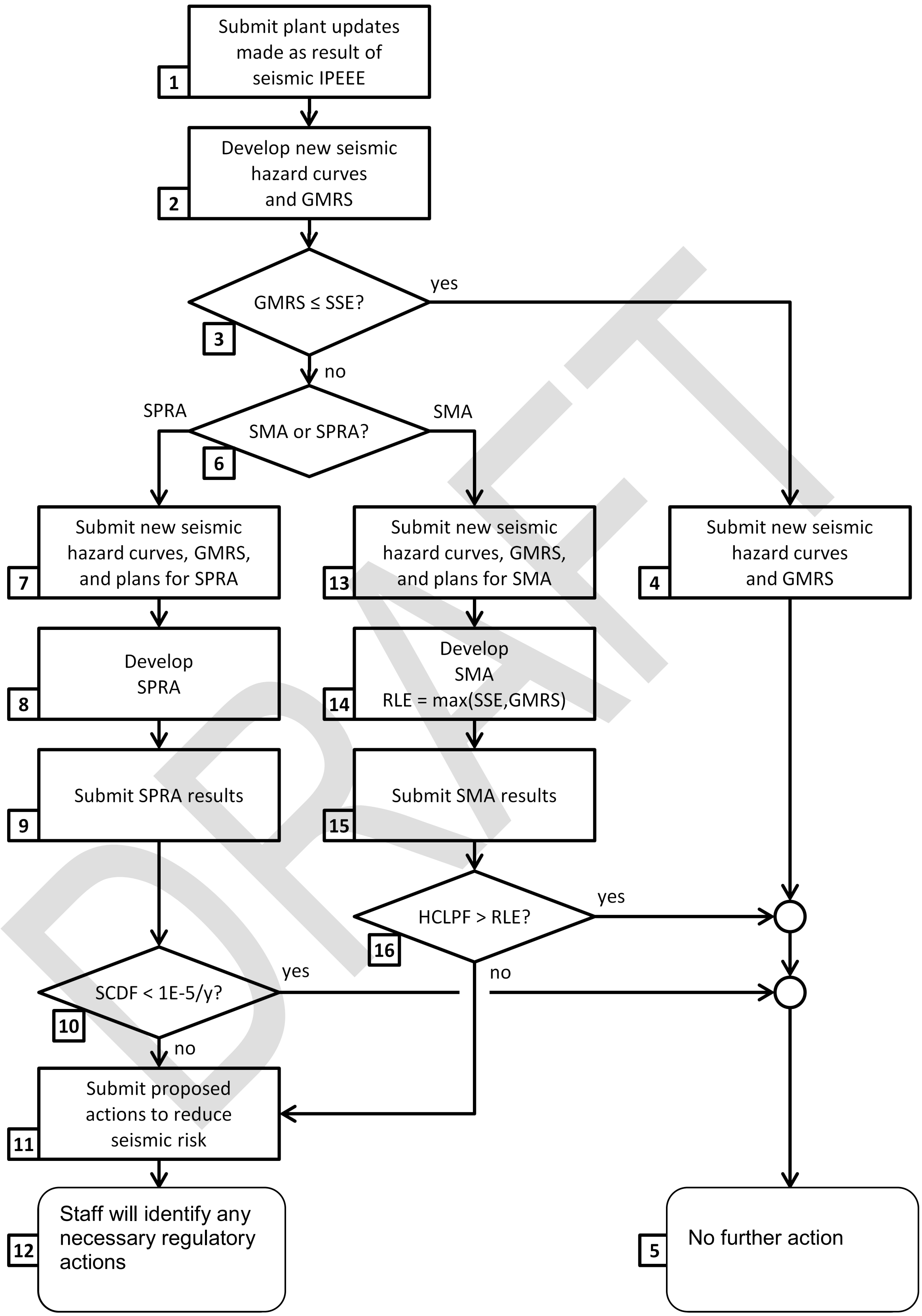

Figure 1. Development of Requested Information and Its Use in Regulatory Analysis.