ORAL ARGUMENT NOT SET

Nos. 14-1210, 14-1212, 14-1216,
14-1217
(consolidated)

IN THE UNITED STATES COURT OF APPEALS
FOR THE DISTRICT OF COLUMBIA CIRCUIT

STATE OF NEW YORK, et al.

` Petitioners,

v.

NUCLEAR REGULATORY COMMISSION, and the
UNITED STATES OF AMERICA,

Respondents.

# ADDENDUM TO AMICUS CURIAE BRIEF OF THE CALIFORNIA STATE ENERGY RESOURCES CONSERVATION AND DEVELOPMENT COMMISSION IN SUPPORT OF PETITIONERS STATE OF NEW YORK, ET AL.

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UNITED STATES
**NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

March 12, 2012

All Power Reactor Licensees and
Holders of Construction Permits in
Active or Deferred Status

SUBJECT: REQUEST FOR INFORMATION PURSUANT TO TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS* 50.54(f) REGARDING RECOMMENDATIONS 2.1, 2.3, AND 9.3, OF THE NEAR-TERM TASK FORCE REVIEW OF INSIGHTS FROM THE FUKUSHIMA DAI-ICHI ACCIDENT

This letter is being issued in accordance with the provisions of Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended (the Act), and the U.S. Nuclear Regulatory Commission (NRC or Commission) regulation in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.54(f). Pursuant to these provisions of the Act or this regulation, you are required to provide further information to support the evaluation of the NRC staff recommendations for the Near-Term Task Force (NTTF) review of the accident at the Fukushima Dai-ichi nuclear facility. The review will enable the staff to determine whether the nuclear plant licenses under your responsibility should be modified, suspended, or revoked. For combined license (COL) holders under 10 CFR Part 52, the issues in NTTF Recommendation 2.1 and 2.3 regarding seismic and flooding reevaluations and walkdowns are resolved. Therefore, COL holders are not required to respond to Enclosures 1 through 4 of this letter. Similarly, information requests in Enclosures 3 and 4 are not applicable to holders of construction permits under 10 CFR Part 50. Operating power reactor licensees under 10 CFR Part 50 are required to respond to all of the information requests.

## BACKGROUND

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tōhoku Earthquake and subsequent tsunami, the NRC established the NTTF in response to Commission direction. The NTTF Charter, dated March 30, 2011, tasked the NTTF with conducting a systematic and methodical review of NRC processes and regulations and determining if the agency should make additional improvements to its regulatory system. Ultimately, a comprehensive set of recommendations contained in a report to the Commission (dated July 12, 2011, SECY-11-0093 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML111861807)) was developed using a decision rationale built around the defense-in-depth concept in which each level of defense-in-depth (namely prevention, mitigation, and emergency preparedness (EP)) is critically evaluated for its completeness and effectiveness in performing its safety function.

The current regulatory approach, and the resultant plant capabilities, gave the NTTF and the NRC the confidence to conclude that an accident with consequences similar to the Fukushima accident is unlikely to occur in the United States (U.S.). The NRC concluded that continued plant operation and the continuation of licensing activities did not pose an imminent risk to public health and safety.

On August 19, 2011, following issuance of the NTTF report, the Commission directed the NRC staff in staff requirements memorandum (SRM) for SECY-11-0093 (ADAMS Accession No. ML112310021), in part, to determine which of the recommendations could and should be implemented without unnecessary delay.

On September 9, 2011, the NRC staff provided SECY-11-0124 to the Commission (ADAMS Accession No. ML11245A158). The document identified those actions from the NTTF report that should be taken without unnecessary delay. As part of the October 18, 2011, SRM for SECY-11-0124 (ADAMS Accession No. ML112911571), the Commission approved the staff's proposed actions, including the development of three information requests under 10 CFR 50.54(f). The information collected would be used to support the NRC staff's evaluation of whether further regulatory action was needed in the areas of seismic and flooding design, and EP.

On December 23, 2011, the Consolidated Appropriations Act, Public Law 112-074, was signed into law. Section 402 of the law also requires a reevaluation of licensees' design basis for external hazards, and expands the scope to include other external events, as described below:

> The Nuclear Regulatory Commission shall require reactor licensees to re-evaluate the seismic, tsunami, flooding, and other external hazards at their sites against current applicable Commission requirements and guidance for such licensees as expeditiously as possible, and thereafter when appropriate, as determined by the Commission, and require each licensee to respond to the Commission that the design basis for each reactor meets the requirements of its license, current applicable Commission requirements and guidance for such license. Based upon the evaluations conducted pursuant to this section and other information it deems relevant, the Commission shall require licensees to update the design basis for each reactor, if necessary.

Reevaluation of the design basis with respect to other external events will be requested later as a separate action from this letter. However, licensees are encouraged to consider this when performing the Recommendation 2.3 walkdowns for flooding.

In the context of Recommendation 2.1 of this 10 CFR 50.54(f) letter, the NRC staff definition of vulnerability[1] is broad enough to capture both prevention and mitigation aspects and also include features of protection such as hardware, procedures, temporary measures, and potentially available off-site resources. Such a definition allows both licensees and the NRC staff to assess plant response to a natural hazard event as an integrated system providing consideration for all available resources. Information resulting from such an evaluation will help the staff decide upon the most appropriate regulatory action focusing on the most beneficial safety enhancements.

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[1] For the purpose of this document, plant-specific vulnerabilities are defined as those features important to safety that when subject to an increased demand due to the newly calculated hazard evaluation have not been shown to be capable of performing their intended safety functions.

ACTION

The NRC has concluded that it requires the information requested in the enclosures to this letter to verify the compliance with your plant's design basis and to determine if additional regulatory actions are appropriate. Therefore, you are required, pursuant to Section 182(a) of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), to submit a response to this letter. You must confirm receipt of this letter within 30 days, however, each attachment contains a topic-specific schedule for response. Your response must be written and signed under oath or affirmation.

The NRC has provided information in each enclosure on acceptable approaches for responding to the information requests. Alternate approaches with appropriate justification will be considered.

This request contains information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB) under an expedited clearance, approval number 3150-0211, which expires September 30, 2012. Prior to the expiration date, the NRC will submit the collection to OMB for renewal.

The burden for these information collections is estimated to average 13,300 hours per response, as detailed in Table 1. This estimate includes the time for reviewing instructions, searching existing data sources, gathering data, performing necessary analyses, and completing and reviewing the collection of information. These estimates represent the average level of effort per plant; actual levels of effort may vary depending upon the results of the hazard analyses. Send comments regarding this burden estimate or any other aspect of these information collections, including suggestions for reducing the burden, to the Information Services Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by email to INFOCOLLECTS.RESOURCE@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0211), Office of Management and Budget, Washington, DC 20503.

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

**Table 1 Burden Estimate (hours)**

| | Hazard Evaluation | Risk/Integrated Assessment | Walkdowns | EP Communications | EP Staffing |
|---|---|---|---|---|---|
| Enclosure 1 | 1700 | 3500 | N/A | N/A | N/A |
| Enclosure 2 | 1300 | 2700 | N/A | N/A | N/A |
| Enclosure 3 | N/A | N/A | 2000 | N/A | N/A |
| Enclosure 4 | N/A | N/A | 2000 | N/A | N/A |
| Enclosure 5 | N/A | N/A | N/A | 50 | 50 |

In accordance with 10 CFR 2.390, "public inspections, exemptions, and requests for withholding," a copy of this letter and your response will be made available for inspection and copying at the NRC Website at www.nrc.gov, and/or at the NRC Public Document Room. If you believe that any of the information to be submitted meets the criteria in 10 CFR 2.390 for withholding from public disclosure, you must include sufficient information, as required by the subsection, to support such a determination.

INFORMATION REQUEST JUSTIFICATION

Hazard Reevaluations and Walkdowns

Current NRC regulations and associated regulatory guidance provide a robust regulatory approach for the evaluation of site hazards associated with natural phenomena. However, this framework has evolved over time as new information regarding site hazards and the potential consequence has become available. As a result, the licensing basis, design, and level of protection from natural phenomena differ among the existing operating reactors in the U.S., depending on when the plant was constructed and licensed for operation. Additionally, the assumptions and factors that were considered in determining the level of protection necessary at these sites vary depending on a number of contributing factors. To date, the NRC has not undertaken a comprehensive re-establishment of the design basis for existing plants to reflect the current state of knowledge or current licensing criteria.

Protection from natural phenomena is critical for safe operation of nuclear power plants. Failure to protect structures, systems, and components (SSCs) important to safety from natural phenomena with appropriate safety margins has the potential to result in common-cause failures with significant consequences, as was demonstrated at Fukushima. Additionally, the consequences of an accident from some natural phenomena may be aggravated by a "cliff-edge" effect, in that a small increase in the hazard (e.g., flooding level) may sharply increase the number of SSCs affected.

As the state of knowledge of these hazards has evolved significantly since the licensing of many of the plants within the U.S., and given the demonstrated consequences from Fukushima, it is necessary to confirm the appropriateness of the hazards assumed for U.S. plants and their ability to protect against them.

In accordance with Commission direction, the NRC staff is implementing the following:

A hazard evaluation consistent with Recommendation 2.1 will be implemented in two phases as follows:

- Phase 1: Issue 10 CFR 50.54(f) letters to all licensees to request that they reevaluate the seismic and flooding hazards at their sites using updated seismic and flooding hazard information and present-day regulatory guidance and methodologies and, if necessary, to request they perform a risk evaluation. The evaluations associated with the requested information in this letter do not revise the design basis of the plant. This letter implements Phase 1.

- Phase 2: Based upon the results of Phase 1, the NRC staff will determine whether additional regulatory actions are necessary (e.g., update the design basis and SSCs important to safety) to provide additional protection against the updated hazards.

The NRC staff's goal is to complete Phase 1 and collect sufficient information to make a regulatory decision for most plants within five years. It is anticipated that collection of this information for all plants will take no longer than seven years.

Information collection on hazard protection walkdowns consistent with Recommendation 2.3 will be implemented in a single-phase. The results from these walkdowns are expected to capture any degraded, non-conforming conditions, and cliff-edge effects for flooding so that they are addressed by the licensee's corrective action program and will provide input to Recommendation 2.1. It is anticipated that this effort will be completed within approximately one year.

Emergency Preparedness

Further, if mitigation is not successful in preventing the release of radioactive materials from the plant, EP provides additional defense-in-depth to minimize exposure to radiation to the public. The accident at Fukushima reinforced the need for effective EP, the objective of which is to ensure the capability to implement effective measures to mitigate the consequences of a radiological emergency. The accident at Fukushima highlighted the need to determine and implement the required staff to fill all necessary positions responding to a multi-unit event. Additionally, there is a need to ensure that the communication equipment relied upon to coordinate the event response during a prolonged station blackout can be powered.

The reevaluation and related analysis being conducted under this request are justified by the need to enhance those EP measures that support the prevention or mitigation of core damage and uncontrolled release of radioactive material. The justification in this letter, as well as the background and discussions in each of its enclosures, provide the reasoning and justification for this request. Moreover, the reevaluation and related analysis will serve to meet NRC's obligation under the Consolidated Appropriations Act, for 2012 (*Pub Law 112-74*), Section 402, and also affords licensees the opportunity to inform the NRC regarding safety-related decisions.

If you have any questions on this matter, please contact your NRC licensing Project Manager.

Sincerely,

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Michael R. Johnson, Director
Office of New Reactors

Enclosures:
1. Recommendation 2.1: Seismic
2. Recommendation 2.1: Flooding
3. Recommendation 2.3: Seismic
4. Recommendation 2.3: Flooding
5. Recommendation 9.3: EP
6. Licensees and Holders of Construction Permits

cc: Listserv

RECOMMENDATION 2.1: SEISMIC

## PURPOSE

The U.S. Nuclear Regulatory Commission (NRC or Commission) is issuing this information request for the following purposes:

- To gather information with respect to Near-Term Task Force (NTTF) Recommendation 2.1, as directed by staff requirements memoranda (SRM) associated with SECY-11-0124 and SECY-11-0137, and the Consolidated Appropriations Act, for 2012 (*Pub Law 112-74*), Section 402, to reevaluate seismic hazards at operating reactor sites

- To collect information to facilitate NRC's determination if there is a need to update the design basis and systems, structures, and components (SSCs) important to safety to protect against the updated hazards at operating reactor sites

- To collect information with respect to the resolution of Generic Issue (GI) 199

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(f), addressees are required to submit a written response to this information request.

## BACKGROUND

The SSCs important to safety in operating nuclear power plants are designed either in accordance with, or meet the intent of Appendix A to 10 CFR Part 100 and Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 2. GDC 2 states that SSCs important to safety at nuclear power plants must be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their intended safety functions. The design bases for these SSCs reflect consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area. The design bases also reflect margin to account for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

In response to the accident at the Fukushima Dai-ichi nuclear power plant caused by the March 11, 2011, Tohoku earthquake and subsequent tsunami, the Commission established a NTTF to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. The purpose of this letter is to gather information with respect to NTTF Recommendation 2.1 for seismic hazards. Recommendation 2.1, as amended by the SRMs associated with SECY-11-0124 and SECY-11-0137, instructs the NRC staff to issue requests for information to licensees pursuant to Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). This information request is for licensees and holders of construction permits under 10 CFR Part 50 to reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance. Based upon this information, the NRC staff will determine whether additional regulatory actions are necessary (e.g., update the design basis and SSCs important to safety) to protect against the updated hazards. In developing Recommendation 2.1, the NTTF recognized that the state of knowledge

of seismic hazard within the United States (U.S.) has evolved and the level of conservatism in the determination of the original seismic design bases should be reexamined.

Since the issuance of GDC 2, the NRC has developed new regulations, regulatory guidance, and several regulatory programs aimed at enhancements for previously licensed reactors. These regulatory programs for enhancements are described in Section 4.1.1 of the NTTF Report, "Recommendations for Enhancing Reactor Safety in the 21st Century." Two recent programs are the individual plant examinations of external events (IPEEEs) and 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 programs.

Individual Plant Examination of External Events:

On June 28, 1991, the NRC issued Supplement 4 to Generic Letter (GL) 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," (ADAMS Accession No. ML031150485). GL 88-20, referred to as the IPEEE program, requested 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; high winds, floods, and other external initiating events, including accidents related to transportation or nearby facilities and plant-unique hazards.

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 licenses (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 Central and Eastern United States (CEUS), such as those around Charleston, SC, and New Madrid, MO. These reviews identified higher seismic hazard estimates than previously assumed, which may result in an 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 should be examined under the Generic Issues Program (GIP).

Generic Issue (GI)-199 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 staff completed the initial screening analysis of GI-199 and held a public meeting in February 2008, (ADAMS Accession Nos. ML073400477 and ML080350189) concluding that GI-199 should proceed to the safety/risk assessment stage of the GIP.

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. The staff compared the new seismic hazard data with the earlier evaluations conducted as part of the IPEEE program. 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 safety/risk assessment also concluded that (1) an immediate safety concern did not exist and (2) adequate protection of public health and safety was not challenged as a result of the new information. The NRC staff presented this conclusion at a public meeting held on October 6, 2010 (ADAMS Accession No. ML102950263). Information Notice 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.

For the GI-199 safety/risk assessment, the NRC staff evaluated the potential risk significance of the updated seismic hazards on seismic core damage frequency (SCDF) estimates. 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. However, as described in NUREG-1742, there are limitations associated with utilizing the inherently qualitative insights from the IPEEE submittals

in a quantitative assessment. In particular, 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, whether additional regulatory action is warranted.

**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 50.34(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"
- 10 CFR 100.23, "Geological and Seismic Siting Criteria"

The seismic design bases for currently operating nuclear power plants were either developed in accordance with, or meet the intent of GDC 2 and 10 CFR Part 100, Appendix A. Although the regulatory requirements in Appendix A to 10 CFR Part 100 are fundamentally deterministic, the NRC process for determining the seismic design basis ground motions for new reactor applications after January 10, 1997, as described in 10 CFR 100.23, requires that uncertainties be addressed through an appropriate analysis such as a probabilistic seismic hazard analysis.

**DISCUSSION**

Recommendation 2.1, as amended by the SRMs associated with SECY-11-0124 and SECY-11-0137, instructs the NRC staff to issue requests for licensees to reevaluate the seismic hazards at their sites using present-day NRC requirements and guidance, and identify actions that are planned to address plant-specific vulnerabilities[1] associated with the updated seismic hazards. Recommendation 2.1 for seismic hazards will be implemented in two phases as follows:

- Phase 1: Issue 10 CFR 50.54(f) letters to all licensees to reevaluate the seismic hazard at their sites using updated seismic hazard information and present-day regulatory guidance and methodologies and, if necessary, to perform a risk evaluation.
- Phase 2: If necessary, and based upon the results of Phase 1, determine whether additional regulatory actions are necessary (e.g., update the design basis and SSCs important to safety) to protect against the updated hazards.

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[1] A definition of vulnerability in the context of this enclosure is as follows: Plant-specific vulnerabilities are those features important to safety that when subject to an increased demand due to the newly calculated hazard evaluation have not been shown to be capable of performing their intended safety functions.

To implement NTTF Recommendation 2.1, the staff is utilizing the general process developed for GI-199 as presented in the draft GL for GI-199 (ADAMS Accession No. ML11710783). This process, described in Attachment 1, asks each addressee to provide information about the current hazard and potential risk posed by seismic events using a progressive screening approach. Depending on the comparison between the reevaluated seismic hazard and the current design basis, the result is either no further risk evaluation or the performance of a seismic risk assessment. Risk assessment approaches acceptable to the staff include a seismic probabilistic risk assessment (SPRA), or a seismic margin assessment (SMA).

Present-day NRC requirements and guidance with respect to characterizing seismic hazards use a probabilistic approach in order to develop a risk-informed performance-based ground motion response spectrum (GMRS) for the site. This approach is described in Regulatory Guide (RG) 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion." RG 1.208 recommends the use of the Senior Seismic Hazard Analysis Committee (SSHAC) approach for treatment of expert judgment and quantifying uncertainty in order to develop seismic source and ground motion models for a Probabilistic Seismic Hazard Analysis used to develop the GMRS for a site.

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"). Part 10 of the 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," provides an acceptable approach for determining the technical adequacy of the SMA approach used to respond to this information request. The SMA approach should include both core damage (accident prevention) and large early release (accident mitigation).

The NRC staff recommends that the SPRA approach at least be a Level 1 with an estimate of large early release frequency (LERF). By including containment performance and extending to Level 2 (including LERF) additional mitigation features that may be under consideration can be incorporated into the analyses. One acceptable approach for determining the technical adequacy of the SPRA is described in RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (ADAMS Accession No. ML090410014) and ASME/ANS RA-Sa-2009). Consistent with the NRC's probabilistic risk assessment (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.

**REQUESTED ACTIONS**

Addressees are requested to perform a reevaluation of the seismic hazards at their sites using present-day NRC requirements and guidance to develop a GMRS. Recently, new consensus seismic source models for the CEUS (NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities"), referred to as the Central and Eastern United States Seismic Source Characterization, have been completed using a SSHAC Level 3 process. Addressees whose plants are located in the CEUS will be able to use this new seismic

source model to characterize the hazard for their plants. Addressees whose plants lie in the Western United States (WUS) are requested to develop seismic source and ground motion models to characterize their regional and site-specific seismic hazards. Consistent with current practice for 10 CFR Part 52, new reactor licensing, WUS addressees should perform a SSHAC Level 3 study to develop a probabilistic seismic hazard analysis.

Addressees are requested to submit, along with the hazard evaluation, an interim evaluation and actions planned or taken to address the reevaluated hazard where it exceeds the current design basis.

While the seismic hazard reevaluation is being performed, NRC staff and stakeholders will continue interacting to develop strategies for screening, prioritization, and potential interim actions as well as implementation guidance for the risk evaluation. For plants where the reevaluated hazard exceeds the current design basis, addressees may opt to perform an SPRA. In addition, an SPRA, rather than a SMA, may be necessary for cases where the SMA screening tables are not usable due to a higher reevaluated hazard (i.e., GMRS). For all other plants where the reevaluated hazard exceeds the current design basis, the NRC will provide guidance on when an SMA option can be used. Factors that the staff will consider to determine whether an SPRA or an SMA is appropriate are (1) the extent to which the reevaluated hazard (GMRS) exceeds the current design basis (SSE), (2) the absolute seismic hazard based on an examination of the probabilistic seismic hazard curves for the site, and (3) previous estimates of plant capacity (e.g., IPEEE insights). The priority for the subsequent completion of the risk assessments by the addressees will also be based on the above factors. For example, as part of the GI-199 safety/risk assessment, the NRC staff found that assuming a factor of 1.3 times the SSE, combined with updated seismic hazard curves, distinguished between plants with lower and higher risk estimates.

Along with an assessment of reactor integrity, the NTTF recommended an evaluation of the spent fuel pool (SFP) integrity. The addressee's evaluation should consider all seismically induced failures that can lead to draining of the SFP. The evaluation should consider SFP walls, liner, penetrations (cooling water supplies or returns, drains), transfer gates and seals, seals and bellows between the SFP, transfer canal, and reactor cavity, sloshing effects (including loss of SFP inventory, wave-induced failures of gates, and subsequent flooding), siphon effects caused by cooling water pipe breaks, and other relevant effects that could lead to a significant loss of inventory of the SFP.

**REQUESTED INFORMATION**

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

<u>Seismic Hazard Evaluation</u>

(1) site-specific hazard curves (common fractiles and mean) over a range of spectral frequencies and annual exceedance frequencies

(2) site-specific, performance-based GMRS developed from the new site-specific seismic hazard curves at the control point elevation(s)

(3) SSE ground motion values including specification of the control point elevation(s)

(4) comparison of the GMRS and SSE (if the GMRS is completely bounded by the SSE, an interim action plan or a risk evaluation is not necessary. However, if the GMRS exceeds the SSE only at higher frequencies information related to the functionality of high-frequency sensitive SSCs is requested. Attachment 1 provides further details)

(5) additional information such as insights from NTTF Recommendation 2.3 walkdown and estimates of plant seismic capacity developed from previous risk assessments to inform NRC screening and prioritization

(6) interim evaluation and actions taken or planned to address the higher seismic hazard relative to the design basis, as appropriate, prior to completion of the risk evaluation described below

(7) selected risk evaluation approach (if necessary)

<u>Seismic Risk Evaluation</u>

(8) SMA or SPRA (depending on criteria discussed above)

A. For plants that perform a SMA, the following information is requested:

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

(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 the values

(4) general bases for screening SSCs

(5) description of the SMA, including the development of its logic models, the seismic response analysis, the results of the evaluation of containment performance, 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) description of the process used to ensure that the SMA is technically adequate, including the dates and findings of peer reviews

(7) identified plant-specific vulnerabilities and actions planned or taken

B. For plants that perform a SPRA, the following information is requested:

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

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

i. 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), and the source of information

iii. seismic fragility parameters

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

v. 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

vi. assumptions about containment performance

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

(4) identified plant-specific vulnerabilities and actions that are planned or taken

(9) SFP Evaluation

A. description of the procedures used to evaluate the SFP integrity

B. results of the evaluation

C. identified actions that have been taken or that will be taken to address vulnerabilities associated with the SFP integrity

**REQUIRED RESPONSE**

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

1. Within 60 days of the date of the NRC's issuance of guidance on screening and prioritization criteria, and the implementation details of the risk assessment, each addressee is requested to submit: (1) its intention to follow the NRC-developed guidance[2], or (2) an alternative approach, including acceptance criteria.

2. Within 1.5 years of the date of this information request, each CEUS addressee is requested to submit a written response consistent with the requested information, seismic hazard evaluation, items 1 through 7 above. Within approximately 30 days of receipt of the last addressee submittal, the NRC staff will have determined the acceptability of the licensee's proposed risk evaluation approach, if necessary, and priority for completion.

3. Within 3 years of the date of this information request, each WUS addressee is requested to submit a written response consistent with the requested information, seismic hazard evaluation, items 1 through 7 above. Within approximately 30 days of receipt of the last addressee submittal, the NRC staff will have determined the acceptability of the licensee's proposed risk evaluation approach, if necessary, and priority for completion.

4. For hazard reevaluations that the NRC determines demonstrate the need for a higher priority, addressees are requested to complete the risk evaluation (items 8B and 9 above) over a period not to exceed 3 years from the date of the prioritization.

5. For hazard reevaluations that the NRC determines do not demonstrate the need for a higher priority, addressees are requested to complete the risk evaluation (items 8A or 8B and 9 above) over a period not to exceed 4 years from the date of the prioritization.

If an addressee cannot meet the requested response date, the addressee must provide a response within 90 days of the date of this information request 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.

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[2] The NRC staff will develop screening and prioritization criteria, and the implementation details of the risk assessment, including criteria for identifying vulnerabilities. This information is scheduled to be developed by November 30, 2012 and the NRC staff will interact with stakeholders, as appropriate during this process.

**Attachment 1 to Seismic Enclosure 1**

## Introduction

This Attachment describes an acceptable process for developing the information requested by the U.S. Nuclear Regulatory Commission (NRC). Figure 1 illustrates the process, which is based on a progressive screening approach. The following paragraphs provide additional discussion about each individual step in Figure 1.

**Step 1.** Addressees should develop site-specific base rock and control point elevation hazard curves (i.e., corresponding to fractile levels of 0.05, 0.16, 0.50, 0.84, and 0.95 and the mean) over a range of spectral frequencies (0.5 Hz, 1 Hz, 2.5 Hz, 5 Hz, 10 Hz, and 25 Hz and peak ground acceleration - PGA) and annual exceedance frequencies ($1 \times 10^{-6}$ and higher) determined from a probabilistic seismic hazard analysis (PSHA) as follows:

- Addressees of plants located in the Central and Eastern United States (CEUS) are expected to use the CEUS Seismic Source Characterization (CEUS-SSC) model (NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities") 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.

- Addressees of plants located in the Western United States (Columbia, Diablo Canyon, Palo Verde, and San Onofre) should 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. Consistent with Regulatory Guide (RG) 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," addressees should use a Senior Seismic Hazard Analysis Committee (SSHAC) study, as described in NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts." Consistent with current practice, as described in NUREG-2117, "Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies," a SSHAC Level 3 study should be performed.

- To remove non-damaging lower-magnitude earthquakes, addressees should either use a lower bound magnitude cutoff of moment magnitude ($M_w$) 5 or the cumulative absolute velocity (CAV) filter for the PSHA. The CAV filter should be limited to $M_w$ less than or equal to 5.5.

- Addressees should use 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." The dynamic site response should be determined through analyses based on either time history or random vibration theory. The subsurface site response model, for both soil and rock sites, should extend to sufficient depth to reach the generic rock conditions as defined in the ground motion models used in the PSHA. In addition, a randomization procedure should be used that appropriately represents the amount of subsurface information at a

- given site. In addition, the randomization procedure should accommodate the variability in soil depth (including depth to generic rock conditions), shear-wave velocities, layer thicknesses, and strain dependant nonlinear material properties at the site. Generally, at least 60 convolution analyses should be performed to define the mean and standard deviation of the site response. Site amplification curves should be developed over a broad range of annual exceedance frequencies ($1\times10^{-6}$ and higher) to facilitate estimation of seismic core damage frequency.

- Addresses should document the low- and high-frequency controlling earthquakes at frequencies of $10^{-4}$ and $10^{-5}$ per year.

- Addressees should use the site-specific hazard curves to develop a performance-based ground motion response spectrum (GMRS) for the site, using the guidance in RG 1.208. The site-specific GMRS should be determined and clearly specified at the same elevation as the design-basis safe shutdown earthquake (SSE) ground motion assuming a site profile with a free surface above the control point elevation.

**Step 2.** Addressees are requested to provide the new seismic hazard curves, the GMRS, and the SSE in graphical and tabular format. Addressees are also requested to provide soil profiles used in the site response analysis as well as the resulting soil amplification functions.

**Step 3.** If the SSE is greater than or equal to the GMRS at all frequencies between 1 and 10 Hz and at the PGA anchor point, then addressees may terminate the evaluation (Step 4)[3] after providing a confirmation, if necessary, that SSCs, which may be affected by high-frequency ground motion, will maintain their functions important to safety.

**Step 4.** This step demonstrates termination of the process for resolution of NTTF, Recommendation 2.1 for plants whose SSE is greater than the calculated GMRS.

**Step 5.** Based on NRC screening criteria, addressees will be requested to perform a seismic margins analysis (SMA) or a seismic probabilistic risk assessment (SPRA). If addressees perform an SPRA, then they are requested to follow Steps 6a and 7a. If addressees perform an SMA, then they are requested to follow Steps 6b and 7b.

**Step 6a.** It is requested that addressees that perform an SPRA ensure that the SPRA is technically adequate for regulatory decision making and includes an evaluation of containment performance and integrity. RG 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 used to respond to this information request.

**Step 6b.** It is requested that addressees that perform an SMA use a composite spectrum review level earthquake, defined as the maximum of the GMRS and SSE at each spectral frequency. The SMA should also include an evaluation of containment performance and

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[3] For plants with only a high frequency ground motion exceedance (above 10 Hz), the documentation should also include a confirmation that affected plant structures and equipment at various elevations will maintain their functions important to safety at the higher acceleration levels.

integrity. The American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009 provides an acceptable approach for determining the technical adequacy of an SMA used to respond to this information request.

**Step 7a.** Document and submit the results of the SPRA to the NRC for review. The "Requested Information" section in the main body of Enclosure 1 identifies the specific information that is requested. In addition, addresses are requested to submit an evaluation of the SFP integrity.

**Step 7b.** Document and submit the results of the SMA to the NRC for review. The "Requested Information" section in the main body of Enclosure 1 identifies the specific information that is requested. In addition, addresses should submit an evaluation of the SFP integrity.

**Step 8.** Submit plans for actions that evaluate seismic risk contributors. NRC staff, industry, and other stakeholders will continue to interact to develop acceptance criteria in order to identify potential vulnerabilities.

**Step 9.** The information provided in Steps 6 through 8 will be evaluated in Phase 2 to consider any additional regulatory actions.

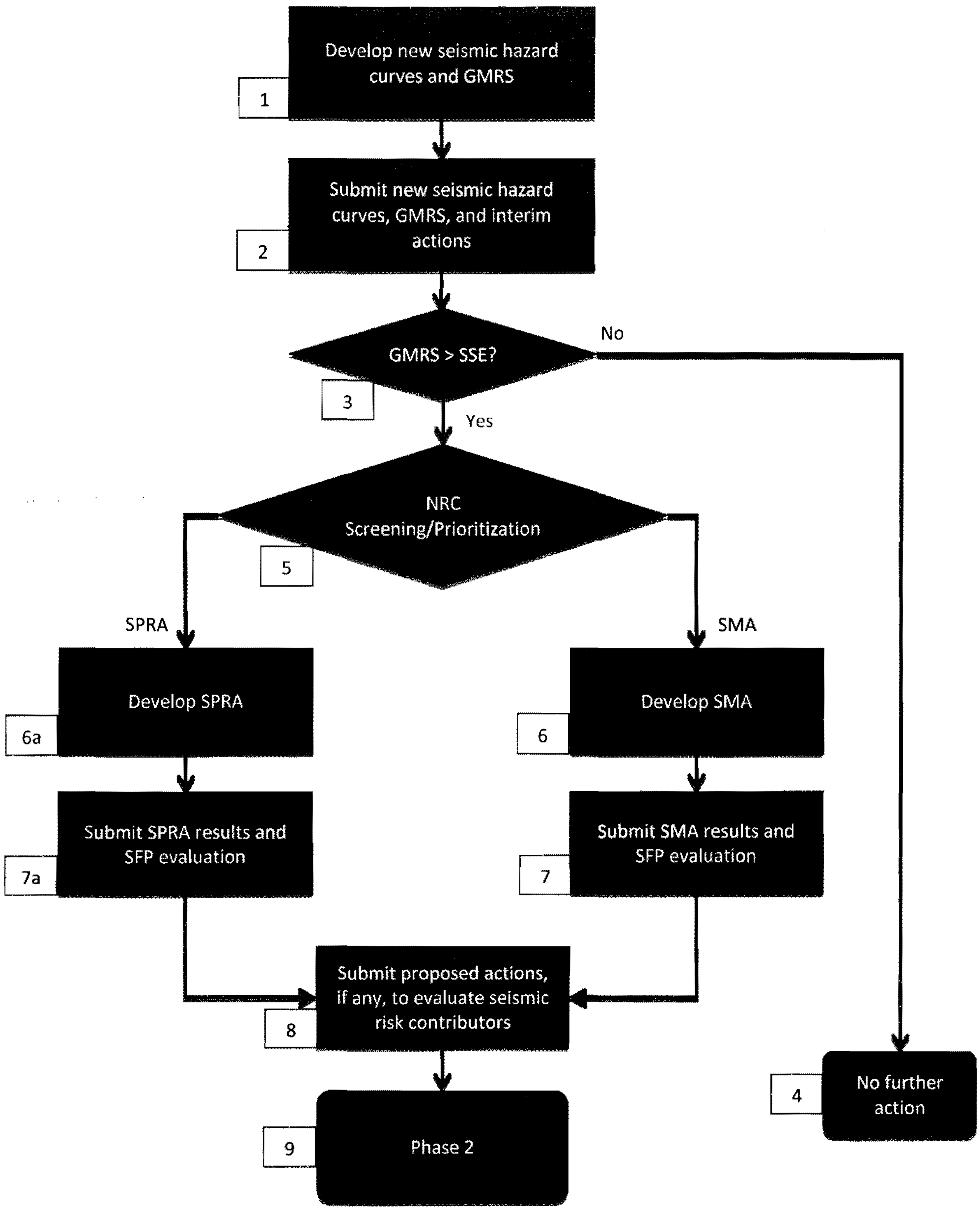


**Figure 1. Development of Requested Information and Its Use in Regulatory Analysis.**

**Enclosure 1 Reference List**

Atomic Energy Act of 1954, as amended, Section 103.b, 161.c, and 182.a

SECY 11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," Agencywide Documents Access and Management System (ADAMS) Accession No. ML11272A111, October 3, 2011.

SECY 11-0124, "Recommended Action to be taken without Delay from the Near-Term Task Force Report," ADAMS Accession No. ML11245A158, September 9, 2011.

10 CFR 50.54(f) – "Conditions of Licenses"

"Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Dai-ichi Accident," ADAMS Accession No. ML111861807, July 12, 2011.

"Generic Issue (GI)-199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,'" ADAMS Accession No. ML051600272, June 9, 2005.

"NRC Generic Letter 1988-020, Supplement 4: Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f)," ADAMS Accession No. ML031150485, June 28, 1991.

NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program – Final Report," ADAMS Accession Nos. ML021270070 and ML021270674, April 2002.

"Identification of a Generic Seismic Issue," ADAMS Accession No. ML051450456, May 26, 2005.

"Results of Initial Screening of Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants.'" ADAMS Accession No. ML073400477, February 1, 2008.

"02/06/2008 Summary of Category 2 Public Meeting with the Public and Industry to Discuss Generic Issue 199, 'Implications of Updated Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,'" ADAMS Accession No. ML080350189, February 8, 2008.

"Results of Safety/Risk Assessment of Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,'" ADAMS Accession No. ML100270582, September 2, 2010.

"10/6/201 – Public Meeting Summary on "Safety/Risk Assessment Results for Generic Issue 199, 'Implication of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,'" ADAMS Accession No. ML102950263, October 29, 2010.

"NRC Information Notice 2010-018: Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,'" ADAMS Accession No. ML101970221, September 2, 2010.

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 50.34(a)(1), (a)(3), (a)(4), (b)(1), (b)(2), and (b)(4), "Contents of Applications; technical information."

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

"Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (Volume 60, page 42622, of the *Federal Register* (60 FR 42622)).

NUREG/BR-0058 Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," ADAMS Accession No. ML042820192, September 30, 2004.

"Draft NRC Generic Letter 2011-XX: Seismic Risk Evaluations for Operating Reactors," ADAMS Accession No. ML111710783, July 26, 2011.

NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," ADAMS Accession No. ML090500182, August 1985.

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,"

Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, ADAMS Accession No. ML090410014, March 2009.

Electric Power Research Institute (EPRI), "CEUS Ground Motion Project Final Report," EPRI Technical Report 1009684, December 2004.

Electric Power Research Institute (EPRI), "Program on Technology Innovation: Truncation of the Lognormal Distribution and Value of the Standard Deviation for Ground Motion Models in the Central and Eastern United States," Technical Report 1014381, Palo Alto, California, August 2006.

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," 2009.

NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," ADAMS Accession Nos. ML080090003 and ML080090004, April 30, 1997.

NUREG-2117, "Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies"

NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines," ADAMS Accession No. ML013100232, October 2001.

Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," ADAMS Accession No. ML070310619, March 11, 2007.

NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities"

RECOMMENDATION 2.1: FLOODING

**PURPOSE**

The U.S. Nuclear Regulatory Commission (NRC or Commission) is issuing this information request for the following purposes:

- To gather information with respect to Near-Term Task Force (NTTF) Recommendation 2.1, as amended by staff requirements memoranda (SRM) associated with SECY-11-0124 and SECY-11-0137, and the Consolidated Appropriations Act, for 2012 (*Pub Law 112-74*), Section 402, to reevaluate seismic and flooding hazards at operating reactor sites

- To collect information to facilitate NRC's determination if there is a need to update the design basis and systems, structures, and components (SSCs) important to safety to protect against the updated hazards at operating reactor sites

- To collect information to address Generic Issue (GI) 204 regarding flooding of nuclear power plant sites following upstream dam failures

Pursuant to Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended, and Title 10 of the *Code of Federal* Regulations (10 CFR), Section 50.54(f), addressees are required to submit a written response to this information request.

**BACKGROUND**

The SSCs important to safety in operating nuclear power plants are designed either in accordance with, or meet the intent of Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 2. GDC 2 states that SSCs important to safety at nuclear power plants must be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their intended safety functions. The design bases for these SSCs reflect consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area. The design bases also reflect margin to account for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

In response to the accident at the Fukushima Dai-ichi nuclear power plant caused by the March 11, 2011, Tohoku earthquake and subsequent tsunami, the Commission established the NTTF to conduct a systematic review of NRC processes and regulations, and to make recommendations to the Commission for its policy direction. The NTTF developed a set of recommendations that are intended to clarify and strengthen the regulatory framework for protection against natural phenomena. The purpose of this letter is to gather information related to NTTF Recommendation 2.1 for flooding hazards. Recommendation 2.1, as amended by the SRMs associated with SECY-11-0124 and SECY-11-0137, instructs the NRC staff to issue requests for information to licensees pursuant to Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). This letter requests licensees and holders of construction permits under 10 CFR Part 50 to reevaluate the flooding hazards at their sites against present-day regulatory guidance and methodologies being used for early site permits and combined license reviews (SECY-11-0124, Staff Recommendations 2 and 4 for

NTTF Recommendation 2.1). This request is consistent with and required by the Consolidated Appropriations Act for 2012 (*Pub Law 112-74*), Section 402.

In developing Recommendation 2.1, the NTTF recognized that, "since the establishment of GDC 2, the NRC's requirements and guidance for protection from seismic events, floods, and other natural phenomena has continued to evolve," and that "as a result, significant differences may exist between plants in the way they protect against design-basis natural phenomena and the safety margin provided."

Since the issuance of GDC 2 in 1971, the NRC has developed new regulations, regulatory guidance, and several regulatory programs aimed at enhancements for previously licensed reactors. A summary of these regulatory programs for enhancements are described in Section 4.1.1 of the NTTF report. From this summary, items of note with regard to flooding include the individual plant examination of external events (IPEEE) program, the new requirement in 10 CFR 100.20 for applications after January 10, 1997, and efforts underway to update Regulatory Guide (RG) 1.59, "Design Basis Floods for Nuclear Power Plants."

<u>Individual Plant Examination of External Events</u>:

On June 28, 1991, the NRC issued Supplement 4 to Generic Letter (GL) 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," (Agencywide Documents Access and Management System (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; high winds, floods, and other external initiating events, including accidents related to transportation or nearby facilities, and plant-unique hazards.

In most cases, licensees used a qualitative progressive-screening approach in lieu of a more quantitative approach to assess the flooding hazard. NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," volumes 1 and 2 issued April, 2002 (ADAMS Accession Nos. ML021270070 and ML021270674) states that "given the substantial uncertainties involved in developing site-specific flood hazard curves, a consideration of possible combinations of multiple effects causing a range of flood levels would have enhanced the robustness of some of the licensee's analyses and lent greater confidence

to their findings." It should be noted that the term "vulnerability" was not defined in GL 88-20. Instead, GL 88-20 states that licensees should provide a discussion on how vulnerability is defined for each external event evaluated. NUREG-1742 notes that "as a result, the use of the term vulnerability varied widely among the IPEEE submittals...Some licensees avoided the term altogether, other stated that no vulnerabilities existed at their plant without defining the word, and still others provided a definition of vulnerability along with a discussion of their findings."

<u>New Requirements for Evaluation of Dam Hazards in 10 CFR 100.20</u>:

The staff established a new requirement in 10 CFR 100.20, "Factors to be Considered when Evaluating Sites," in 1996. The requirement in 10 CFR 100.20(b) states that for applications submitted on or after January 10, 1997, the nature and proximity of man-related hazards must be evaluated to establish site parameters for use in determining whether a plant design can accommodate commonly occurring hazards, and whether the risk of other hazards is very low. A parenthetical statement in the new regulation specifically identifies dams as hazards to be evaluated at a plant site.

<u>Tsunami and Regulatory Guide 1.59 Updates</u>:

Following the Sumatra earthquake and its accompanying tsunami in December 2004, the NRC staff initiated a study to examine tsunami hazards at power plant sites. Study results are documented in NUREG/CR-6966, "Tsunami Hazard Assessment at Nuclear Power Plant Sites in the United States of America," which was published in March 2009. As the NTTF report notes, "while tsunami hazards are not expected to be the limiting flood hazard for operating plants sited on the Atlantic Ocean and the Gulf of Mexico, plants in these coastal regions do not currently include an analysis of tsunami hazards in their licensing basis."

Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," was originally issued in 1973. The most recent version is Revision 2, published in 1977, including an errata dated July 1980, and a substitution of methods presented in Appendix A (ADAMS Accession No. ML003740388). NRC staff is in the process of updating RG 1.59 to address advances in flooding analysis in the 35 years since Revision 2 was published. Although the update to RG 1.59 update is not complete, NUREG/CR7046, "Design Basis Flood Estimation for Site Characterization at Nuclear Power Plants in the United States of America," was published in November 2011. This report documents present-day methodologies used by the NRC to review early site permits (ESPs) and combined license (COL) applications.

<u>GI-204: Flooding of Nuclear Power Plant Sites following Upstream Dam Failures</u>:

Page 28 of the NTTF report states that, "In August 2010, the NRC initiated a proposed GI regarding flooding of nuclear power plant sites following upstream dam failures." The NRC staff approved this generic issue as GI-204 on February 29, 2012. The staff notes that the flood hazard information gathered by this 10 CFR 50.54(f) request will be applicable to the resolution of GI-204.

**APPLICABLE REGULATORY REQUIREMENTS**

- 10 CFR 50.34(a)(1), (a)(3), (a)(4), (b)(1), (b)(2), and (b)(4)
- 10 CFR 50.54, "Conditions of Licenses"
- Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, GDC 2, "Design Bases for Protection against Natural Phenomena"
- Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100
- Subpart B, "Evaluation Factors for Stationary Power Reactors Site Applications On or After January 10, 1997," to 10 CFR Part 100

In GDC 2 it states that SSCs important to safety at nuclear power plants must be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their intended safety functions. The design bases for these SSCs are to reflect appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area. The design bases are also to reflect sufficient margin to account for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

Present-day regulations for reactor site criteria (Subpart B to 10 CFR Part 100 for applications on or after January 10, 1997) states, in part, that the physical characteristics of the site, including hydrology, must be evaluated and site parameters established such that potential threats from such physical characteristics will pose no undue risk to the type of facility proposed to be located at the site (10 CFR 100.21(d)). Factors to be considered when evaluating sites includes the nature and proximity of dams and other man-related hazards (10 CFR 100.20(b)) and the physical characteristics of the site, including the hydrology (10 CFR 100.20(c)).

**DISCUSSION**

The NTTF recommended that the Commission direct several actions to ensure adequate protection from natural phenomena, consistent with the current state of knowledge and analytical methods. These actions should be undertaken to prevent fuel damage and to ensure containment and spent fuel pool integrity. In particular, Recommendation 2.1 states, "Order licensees to reevaluate the seismic and flooding hazards at their sites against current NRC requirements and guidance, and if necessary, update the design basis and SSCs important to safety to protect against the updated hazards."

Staff assessment of Recommendation 2.1 is discussed in SECY-11-0124. Staff noted that the assumptions and factors that were considered in flood protection at operating plants vary. In some cases, the design bases did not consider the effects from local-intense precipitation and related site drainage. In other cases, the probable maximum flood is calculated differently at units co-located at the same site, depending on the time of licensing, resulting in different

design-basis flood protection. The NTTF and the staff noted that some plants rely on operator actions and temporary flood mitigation measures such as sandbagging, temporary flood walls and barriers, and portable equipment to perform safety functions. For several sites, the staff noted that not all appropriate flooding hazards are documented in the updated final safety analysis report. The NTTF and the staff also noted that flooding risks are of concern because of a "cliff-edge" effect, in that the safety consequences of a flooding event may increase sharply with a small increase in the flooding level. Therefore, the staff concluded that all licensees should confirm that SSCs important to safety are adequately protected from flooding hazards.

In the SRM to SECY-11-0124 the Commission approved the staff's proposed actions, which were to implement the NTTF recommendations as described in the SECY without delay. With regard to reevaluating flooding hazards, staff's approved actions are to:

1. Initiate stakeholder interactions to discuss application of present-day regulatory guidance and methodologies being used for ESP and COL reviews to the reevaluation of flooding hazards at operating reactors.

2. Develop and issue a request for information to licensees pursuant to 10 CFR 50.54(f) to:

    a. reevaluate site-specific flooding hazards using the methodology discussed in Item 1 above, and

    b. identify actions that have been taken or are planned to address plant-specific vulnerabilities associated with the updated flooding hazards.

The SRM to SECY-11-0124 also directed the NRC staff to do the following:

- For Recommendation 2.1, when the staff issues the requests for information to licensees pursuant to 10 CFR 50.54(f) to identify actions that have been taken or are planned to address plant-specific vulnerabilities associated with the reevaluation of seismic and flooding hazards, the staff should explain the meaning of "vulnerability."

- The staff should inform the Commission, either through an Information Paper or briefing of the Commissioners' Assistants, when it has developed the technical bases and acceptance criteria for implementing Recommendation 2.1, 2.3, and 9.3.

Additionally, the Consolidated Appropriations Act, for 2012 (*Pub Law 112-74*), Section 402, directs the NRC to "require reactor licensees to reevaluate the seismic, tsunami, flooding, and other external hazards at their sites against current applicable Commission requirements and guidance for such licensees as expeditiously as possible, and thereafter, when appropriate, as determined by the Commission, and require each licensee to respond to the Commission that the design basis for each reactor meets the requirements of its license, current applicable Commission requirements and guidance for such license." These other external hazards can include meteorological and other natural phenomena that could reduce or limit the capacity of safety-related cooling water supplies. These other external hazards will be addressed separately from this information request.

Following the Commission's direction to implement the staff's proposed actions without delay, the NRC staff will implement Recommendation 2.1 in two phases, as follows:

- Phase 1: Issue 10 CFR 50.54(f) letters to all licensees to reevaluate the seismic and flooding hazards at their sites against present-day regulatory guidance and methodologies used for ESP and COL reviews.

- Phase 2: If necessary, and based upon the results of Phase 1, determine whether additional regulatory actions are necessary (e.g., update the design basis and SSCs important to safety) to protect against the updated hazards

This information request addresses only Phase 1; Phase 2 will be conducted after receiving responses to this request.

The NRC staff will interact with industry and stakeholders to develop approaches that can be applied in a uniform and consistent manner across the different sites and plant conditions. This type of an integrated approach will allow the NRC and industry time to assess the significance of any new information related to the hazard evaluation in a systematic manner. This approach is also consistent with Commission direction to initiate stakeholder interactions. As such, responses to this request for information are expected in stages, as outlined in the Required Response section.

Because of the experience gained by both the NRC and the industry in preparing and reviewing numerous ESPs and COLs, present-day methodologies associated with evaluating flooding hazards at plant sites are well documented. It is anticipated that some interactions will be required with the industry and other stakeholders on particulars associated with implementing these methodologies for the existing plants (e.g., certain data collection activities are likely to be needed). However, the timeframe outlined in the requested response section takes this into account. General steps to develop the flooding hazard evaluation are discussed under the requested actions section below, and detailed steps are provided in Attachment 1.

Information related to the identification of actions that will be taken or planned to be taken to address plant-specific vulnerabilities will inform staff's development of "acceptance criteria" necessary to conduct Phase 2, or to address other regulatory actions as necessary. The approaches and methodology used to develop this information requires multiple interactions between the NRC staff, industry, and other stakeholders. The timeframe discussed in the requested response section explicitly recognizes this aspect.

**REQUESTED ACTIONS**

Addressees are requested to perform a reevaluation of all appropriate external flooding sources, including the effects from local intense precipitation on the site, probable maximum flood (PMF) on stream and rivers, storm surges, seiches, tsunami, and dam failures. It is requested that the reevaluation apply present-day regulatory guidance and methodologies being used for ESP and COL reviews including current techniques, software, and methods used in present-day standard engineering practice to develop the flood hazard. The requested information will be gathered in

Phase 1 of the NRC staff's two phase process to implement Recommendation 2.1, and will be used to identify potential vulnerabilities[1].

For the sites where the reevaluated flood exceeds the design basis, addressees are requested to submit an interim action plan that documents actions planned or taken to address the reevaluated hazard with the hazard evaluation.

Subsequently, addressees should perform an integrated assessment of the plant to identify vulnerabilities and actions to address them. The scope of the integrated assessment report will include full power operations and other plant configurations that could be susceptible due to the status of the flood protection features. The scope also includes those features of the ultimate heat sinks (UHS) that could be adversely affected by the flood conditions and lead to degradation of the flood protection (the loss of UHS from non-flood associated causes are not included). It is also requested that the integrated assessment address the entire duration of the flood conditions.

**REQUESTED INFORMATION**

The NRC staff requests that each addressee provide the following information. Attachment 1 provides additional information regarding present-day methodologies and guidance used by the NRC staff performing ESP and COL reviews. The attachment also provides a stepwise approach for assessing the flood hazard that should be applied to evaluate the potential hazard from flood causing mechanisms at each licensed reactor site.

1. **Hazard Reevaluation Report**

   Perform a flood hazard reevaluation. Provide a final report documenting results, as well as pertinent site information and detailed analysis. The final report should contain the following:

   a. Site information related to the flood hazard. Relevant SSCs important to safety and the UHS are included in the scope of this reevaluation, and pertinent data concerning these SSCs should be included. Other relevant site data includes the following:

      i. detailed site information (both designed and as-built), including present-day site layout, elevation of pertinent SSCs important to safety, site topography, as well as pertinent spatial and temporal data sets
      ii. current design basis flood elevations for all flood causing mechanisms
      iii. flood-related changes to the licensing basis and any flood protection changes (including mitigation) since license issuance
      iv. changes to the watershed and local area since license issuance

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[1] A definition of vulnerability in the context of this enclosure is as follows: Plant-specific vulnerabilities are those features important to safety that when subject to an increased demand due to the newly calculated hazard evaluation have not been shown to be capable of performing their intended functions.

v. current licensing basis flood protection and pertinent flood mitigation features at the site
vi. additional site details, as necessary, to assess the flood hazard (i.e., bathymetry, walkdown results, etc.)

b. Evaluation of the flood hazard for each flood causing mechanism, based on present-day methodologies and regulatory guidance. Provide an analysis of each flood causing mechanism that may impact the site including local intense precipitation and site drainage, flooding in streams and rivers, dam breaches and failures, storm surge and seiche, tsunami, channel migration or diversion, and combined effects. Mechanisms that are not applicable at the site may be screened-out; however, a justification should be provided. Provide a basis for inputs and assumptions, methodologies and models used including input and output files, and other pertinent data.

c. Comparison of current and reevaluated flood causing mechanisms at the site. Provide an assessment of the current design basis flood elevation to the reevaluated flood elevation for each flood causing mechanism. Include how the findings from Enclosure 4 of this letter (i.e., Recommendation 2.3 flooding walkdowns) support this determination. If the current design basis flood bounds the reevaluated hazard for all flood causing mechanisms, include how this finding was determined.

d. Interim evaluation and actions taken or planned to address any higher flooding hazards relative to the design basis, prior to completion of the integrated assessment described below, if necessary.

e. Additional actions beyond Requested Information item 1.d taken or planned to address flooding hazards, if any.

**2. Integrated Assessment Report**

For the plants where the current design basis floods do not bound the reevaluated hazard for all flood causing mechanisms, provide the following:

a. Description of the integrated procedure used to evaluate integrity of the plant for the entire duration of flood conditions at the site.

b. Results of the plant evaluations describing the controlling flood mechanisms and its effects, and how the available or planned measures will provide effective protection and mitigation. Discuss whether there is margin beyond the postulated scenarios.

c. Description of any additional protection and/or mitigation features that were installed or are planned, including those installed during course of reevaluating

the hazard. The description should include the specific features and their functions.

d. identify other actions that have been taken or are planned to address plant-specific vulnerabilities.

**REQUIRED RESPONSE**

Within approximately 60 days of the date of this information request, NRC staff will determine the priority for each reactor site to complete the hazard reevaluation report. The site priority will determine the submittal date for addressees to provide written responses to Requested Information item 1 (Hazard Reevaluation Report).

In accordance with Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), an addressee must respond as described below.

1. Within 60 days of the date of the NRC's issuance of guidance on implementation details of the Integrated Assessment Report, including criteria for identifying vulnerabilities, submit an approach for developing an Integrated Assessment Report including criteria for identifying vulnerabilities[2].

2. In accordance with the NRC's prioritization plan, within 1- to 3-years from the date of this information request, submit the Hazard Reevaluation Report. Include the interim action plan requested in item 1.d, if appropriate.

3. Within 2 years following submittal of the Hazard Reevaluation Report to the NRC, any addressee who is requested to complete an Integrated Assessment should submit written responses to Requested Information item 2.

If an addressee cannot meet the requested response date, the addressee must provide a response within 90 days of the date of this information request 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 prioritization described above will be based on information from COL and ESP applications, updated hazard levels if new information exists, and site-specific circumstances. This prioritization scheme is intended to use both the NRC and industry resources most effectively.

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

[2] The NRC staff will develop the implementation details of the Integrated Assessment Report, including criteria for identifying vulnerabilities This information is scheduled to be developed by November 30, 2012 and the NRC staff will interact with stakeholders, as appropriate during this process.

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.

**Attachment 1 to Recommendation 2.1: Flooding Enclosure 2**

## PROCEDURE

The steps shown in Figure 1 of this attachment represent an acceptable approach to perform the reevaluation of the flood hazard and integrated assessment. The flood hazard reevaluation should address all flood causing mechanisms that are pertinent to the site based on the geographic location and interface of the plant with the hydrosphere. The reason for omitting any of these flood causing mechanisms should be clearly discussed in the final report. A discussion of typical flood causing mechanisms is included below. Many types of flood causing mechanisms are included in that discussion, but it is important to note that each site should address unique characteristics and any additional flood causing mechanisms identified.

**Step 1**:
All licensees should review information concerning the current flooding hazard against that for which the plant is designed. This information will be used in the following steps for reevaluation of the flood hazard. Pertinent information includes, but is not limited to, the following:

- Current design basis flood hazard
- Flood elevations and other effects considered in the flood protection[3] for all flood causing mechanisms.
- Changes in licensing basis since initial licensing including site drainage characteristic and modification, watershed changes, new dam construction, revision of dam operations
- New information pertinent to the hydrologic characteristics including changes to dam operation, new flood studies and changes to meteorological basis (e.g., maximum precipitation studies)
- Pertinent information from site-related or watershed-related studies
- Site changes since issuance of the operating license (new barriers, openings, revised drainage systems, new structures, etc.)
- Flood protection mechanisms and identifying characteristics (e.g., structures and procedures)
- Pertinent features identified in site walkdowns

**Step 2**:
Reevaluate the flood hazard based on present day regulatory guidance and methodologies for each flood causing mechanism. Using any new site-related information and site details identified in Step 1, evaluate all possible flood causing mechanisms. Documentation of all methodologies should be discussed. This step of the process reiterates the current hierarchical hazard assessment (HHA) used by U.S. Nuclear Regulatory Commission (NRC) staff. The HHA is described as a progressively refined, stepwise estimation of the site-specific hazards that evaluates the safety of the site with the most conservative plausible assumptions consistent with available data.

[3] Examples of other effects include dynamic wave effects, scouring, and debris transportation

(a) Select one flood causing mechanism to be reanalyzed
(b) Develop a conservative estimate of the site related parameters using simplifying assumptions for a flood causing mechanism and perform the reevaluation.
(c) Determine if the reevaluated flood hazard elevation (from Step 2b) is higher than the original design flood elevation for the selected flood causing mechanism. If not, use this flood elevation for this causal mechanism in Step 3.
(d) Determine if the site-related parameters can be further refined. If yes, perform reevaluation (repeat step 2c). If no, use this flood elevation for this causal mechanism in Step 3.
(e) Determine if all flood causing mechanisms have been addressed. If yes, continue to Step 3. If no, select another flood causing mechanism (Step 2a).

**Step 3:**
For each flood causing mechanism, compare the final flood elevations from the hazard reevaluation against the current design basis flood elevations. Using this comparison, determine whether the design basis flood bounds each reevaluated hazard from Step 2. If it is determined that the current design basis flood bounds all of the reevaluated hazards, proceed to Step 4. If not all of the reevaluated hazards are bounded by the current design basis flood, proceed to Step 6 for additional analysis.

**Step 4**:
Submit a report in accordance with Requested Information item 1, Hazard Reevaluation Report. It is anticipated that activities associated with the NTTF Recommendation 2.3 are completed and form a partial basis for the information requested.

**Step 5:**
No further action is required. This step demonstrates termination of the process for resolution of NTTF Recommendation 2.1.

**Step 6:** Submit a report in accordance with the Requested Information item 1, Hazard Reevaluation Report, including any relevant information from the results of plant walkdown activities related to NTTF Recommendation 2.3. Also, provide plans for conducting further analysis (steps 7 through 9) and submitting the final report identified in Requested Information item 2.

**Step 7**:
For the flood causing mechanisms that were not bounded, or for a controlling flood causing mechanism, perform an integrated assessment using the procedures developed in interactions with the NRC staff. The purpose of the integrated assessment is to determine the effectiveness of the existing design basis and any other planned or installed features for the protection and mitigation of flood conditions for the entire duration of the flood.

**Step 8:**
Identify vulnerabilities, if any, as a result of the assessment conducted in Step 7. Also, identify any planned actions or actions that were already taken to address these vulnerabilities.

**Step 9:**
Submit a report in accordance with the Requested Information item 2. Include a brief summary of the flood causing mechanisms and the associated parameters that were used in the assessment.

**Step 10**:
The information provided in Step 9 will be evaluated by the NRC in Phase 2 to consider any additional regulatory actions.

**FLOOD CAUSING MECHANISMS**

The NRC regulations require that structure, systems and components (SSCs) important to safety of a nuclear power plant are adequately protected from the adverse effects of flooding. The NRC staff discusses the approach for determining the flood hazard for new reactors in its current guidance documents, NUREG-0800 and NUREG/CR-7046.

As part of analyzing the flood hazard, it is important to list all plausible flood causing mechanisms that are capable of generating a severe flood at the site and to recognize that several scenarios of a particular flood causing mechanism can affect the site. For example, extreme precipitation can cause flooding in adjacent rivers, near-by tributaries, and on-site drainage facilities. Similarly, flood causing mechanisms that are not plausible at a particular site may also be ruled out. Present day NRC staff guidance applies the HHA (see NUREG/CR-7046) to each pertinent flood causing mechanism at a site.

The following is a list of flood causing mechanisms that should be addressed in a flood hazard analysis. Site specific characteristics may warrant review of other mechanisms in addition to those listed here.

1. Local Intense Precipitation

Local intense precipitation is a measure of the extreme precipitation at a given location. Generally, local intense precipitation values are developed using methods called Probable Maximum Precipitation (PMP) based on the methods developed by the federal government and published in hydrometeorological reports (HMR) by the National Weather Service. For extreme precipitation, localized precipitation values are developed using methods in HMR 52 (eastern areas of the United States (U.S.)) as well as regionalized reports within the HMR publication series.

The elevation of the site is not relevant for mitigation of flooding from local intense precipitation. The runoff carrying capacity of the site grading design and the performance of any active or passive drainage systems would determine the depth and velocity of surface runoff at the site. Typically, any active drainage system should be considered non-functional at the time of local intense precipitation event. Generally, runoff losses should be ignored during the local intense precipitation event to maximize the runoff. Hydraulic parameters that affect the depth and velocity of flow should be chosen carefully and should be consistent with values used in standard engineering practice.

2. Flooding in Streams and Rivers

The probable maximum flood (PMF) in rivers and streams adjoining the site should be determined by applying the PMP to the drainage basin in which the site is located. The PMF is based on a translation of PMP rainfall on a watershed to flood flow. The estimation of PMP for regional areas within the U.S. is based on HMRs and the appropriate regional report should be used to develop the PMP for a given site and watershed. The PMP is a deterministic estimate of the theoretical maximum depth of precipitation that can occur at a time of year of a specified area. A rainfall-to-runoff transformation function, as well as runoff characteristics, based on the topographic and drainage system network characteristics and watershed properties are needed to appropriately develop the PMF hydrograph. The PMF hydrograph is a time history of the discharge and serves as the input parameter for other hydraulic models which develop the flow characteristics including flood flow and elevation. The U.S. Army Corps of Engineers hydrologic and hydraulic methods are widely accepted in engineering practice. However, unique characteristics or preference of the analysis may dictate use of other models. Appropriate justification for selection of methods, data and models would depend on site-specific circumstances.

3. Dam Breaches and Failures

Flood waves resulting from the breach of upstream dams, including domino-type or cascading dam failures should be evaluated for the site. Water storage and water control structures (such as onsite cooling or auxiliary water reservoirs and onsite levees) that may be located at or above SSCs important to safety should also be evaluated. Additional effects for earthen embankments, such as sediment, should also be considered. Models and methods used to evaluate the dam failure and the resulting effects should be applicable to the type of failure mechanism and should be appropriately justified. Recent analyses completed by State and Federal agencies with appropriate jurisdiction for dams within the watershed may be used.

4. Storm Surge

Storm surge is the rise of offshore water elevation caused principally by the shear force of the hurricane or tropical depression winds acting on the water surface. Technical reports, from the National Oceanic and Atmospheric Administration (NOAA), provide guidance on developing wind fields for a probable maximum hurricane. The wind field parameter is input to coastal hydrodynamics simulation model that predict water surface rise based on the shear forces imparted by the wind. However, appropriate justification for selection of methods, data, and models depends on site-specific circumstances.

5. Seiche

A seiche is an oscillation of the water surface in an enclosed or semi-enclosed water body initiated by an external cause. If a seiche is determined to be possible at the site, then appropriate numerical modeling may be needed. For bays and lakes with irregular geometries and variable bathymetries, numerical longwave hydrodynamics modeling may be the only viable technique to determine hazard.

6. Tsunami

A tsunami is a series of water waves generated by a rapid, large scale disturbance of a water body due to seismic, landslide or volcanic tsunamingenic sources. An assessment with respect to tsunami can include a stepwise approach addressing: the susceptibility of the site's region subject to tsunami, the susceptibility of the plant site affected by tsunami, and specific hazards of the site posed to safety of the plant by tsunami.

7. Ice Induced Flooding

Ice jams and ice dams can cause flooding by impounding water upstream of a site and subsequently collapsing or downstream of a site impounding and backing up water. There is no method to assess a probable maximum ice jam or ice dam, therefore, historical records are generally accessed to determine the most severe historical event in the vicinity of the site. This method is based on an observed historical observation and reasonable margin should be considered.

8. Channel Migration or Diversion

Flood hazard associated with channel diversion is due to the possible migration either toward the site or away from it. For natural channels adjacent to the site, historical and geomorphic processes should be reviewed for possible tendency to meander. For man-made channels, canals or diversions used for the conveyance of water located at a site, possible failure of these structures should be considered.

9. Combined Effect Flood

For flood hazard associated with combined events, American Nuclear Society (ANS) 2.8-1992 provides guidance for combination of flood causing mechanisms for flood hazard at nuclear power reactor sites. In addition to those listed in the ANS guidance, additional plausible combined events should be considered on a site specific basis and should be based on the impacts of other flood causing mechanisms and the location of the site.

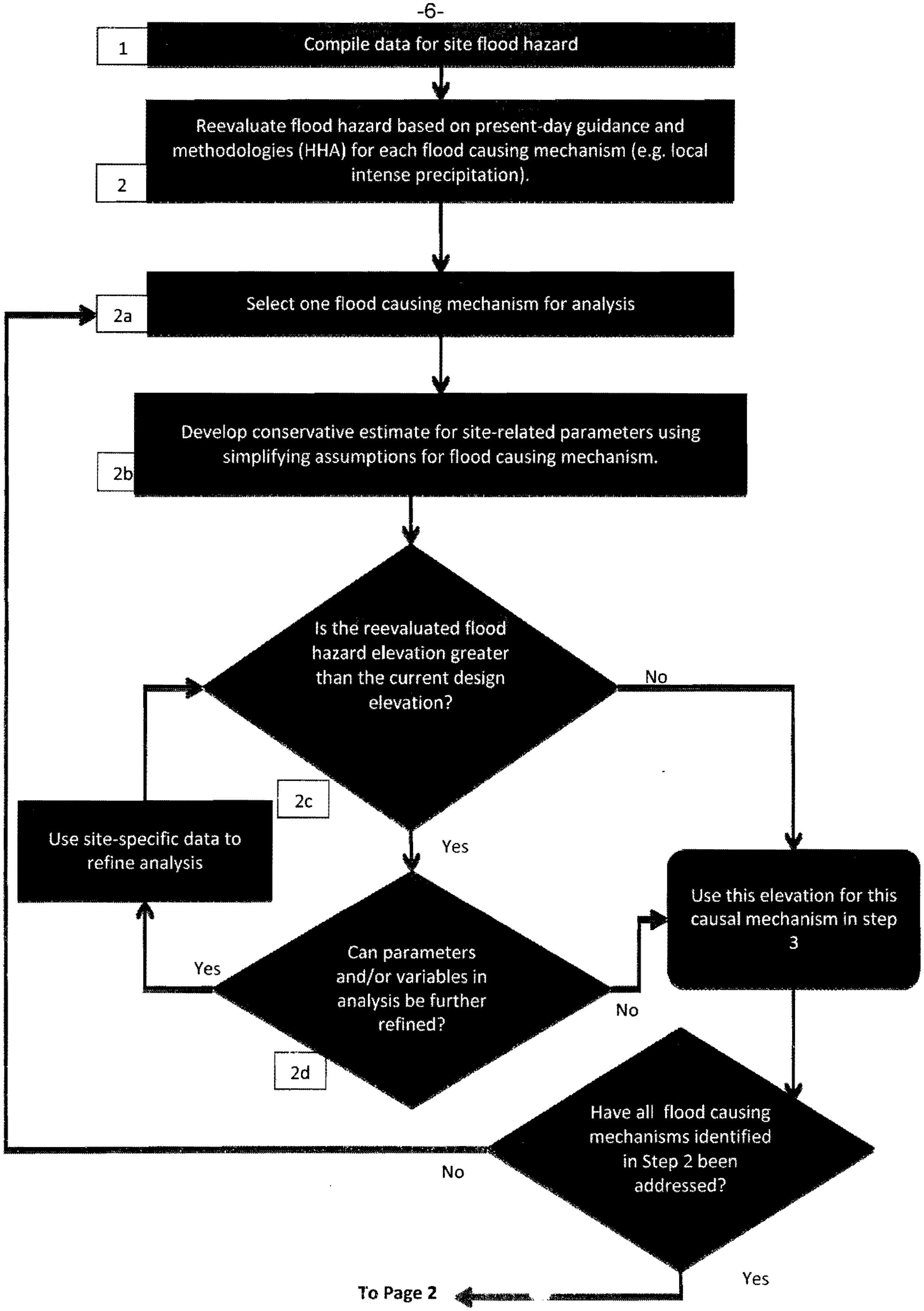


Figure 1. Development of Requested Information and Its Use in Regulatory Analysis. Page 1 of 2

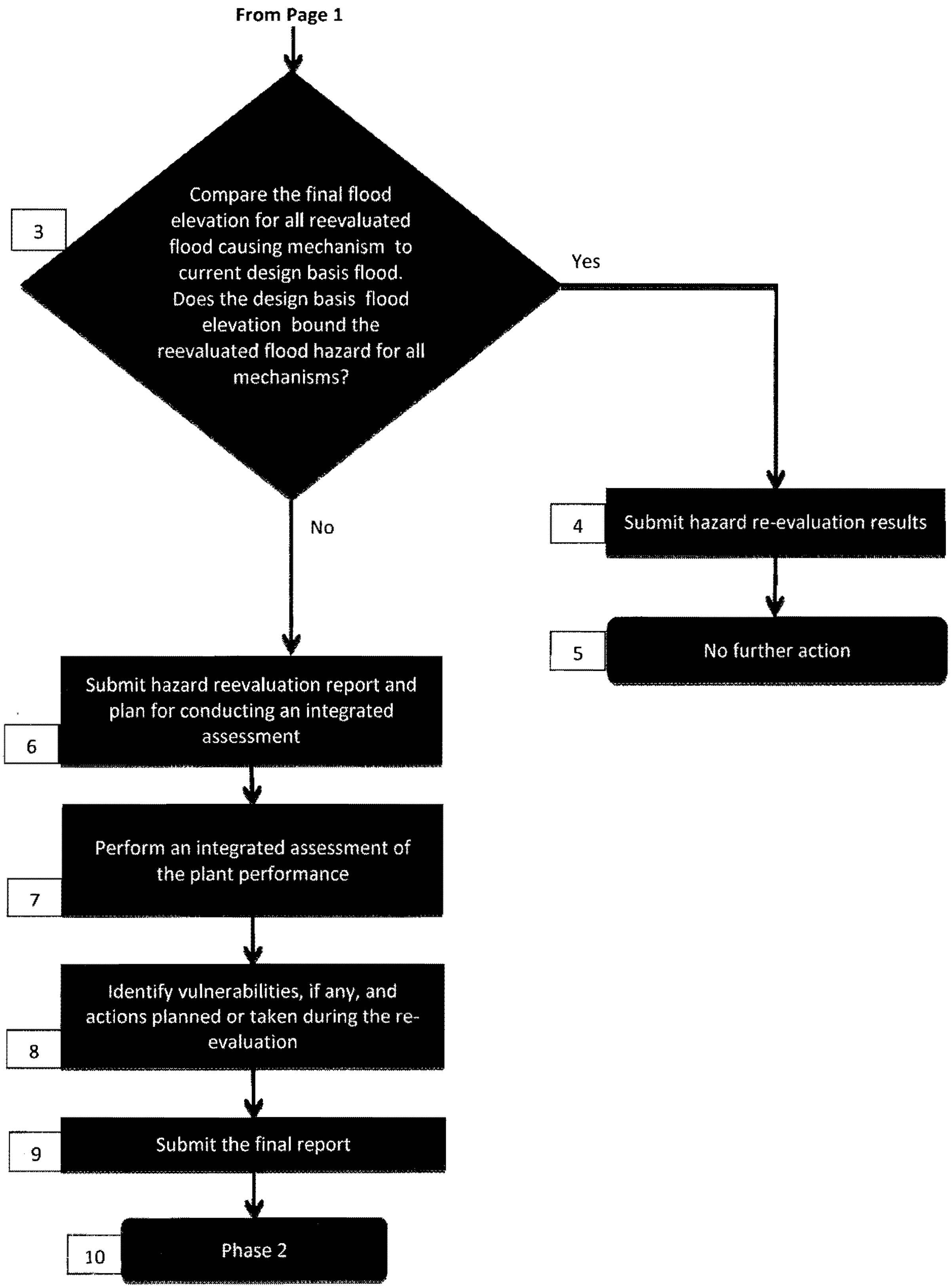


Figure 1. Development of Requested Information and Its Use in Regulatory Analysis. Page 2 of 2

**Enclosure 2 Reference List**

Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended

SECY 11-0124, "Recommended Actions To Be Taken Without Delay from the Near-Term Task Force Report," Agencywide Documents and Management System (ADAMS) Accession No. ML11245A158, September 9, 2011.

SECY 11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," ADAMS Accession No. ML11272A111, October 3, 2011.

"Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Dai-ichi Accident," ADAMS Accession No. ML111861807, July 12, 2011.

10 CFR 50.54(f) – "Conditions of Licenses"

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

Appendix A to 10 CFR Part 50, General Design Criteria 2

"Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (Volume 60, page 42622, of the *Federal Register* (60 FR 42622))

Supplement 4 to GL 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," ADAMS Accession No. ML031150485, June 28, 1991.

10 CFR 100.20, "Factors to be Considered when Evaluating Sites,"

NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," ADAMS Accession No. ML042820192, September 30, 2004.

SRM SECY 11-0124, "Recommended Actions To Be Taken Without Delay from the Near-Term Task Force Report," ADAMS Accession No. ML112911571, October 18, 2011.

SRM SECY 11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," ADAMS Accession No. ML113490055, dated December 15, 2011.

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition – Site Characteristics and Site Parameters (Chapter 2)," ADAMS Accession No. ML070400364, March 2007.

NUREG/CR-7046, PNNL-20091, "Design-Basis Flood Estimation for Site Characterization at Nuclear Power Plants in the United States of America." ADAMS Accession No. ML11321A195, November 2011.

RG 1.29, "Seismic Design Classification," Revision 4, ADAMS Accession No. ML070310052, March 2007.

RG 1.59, "Design Basis Floods for Nuclear Power Plants," Revision 2, ADAMS Accession No. ML003740388, August 1977.

RG 1.102, "Flood Protection for Nuclear Power Plants," Revision 1, ADAMS Accession No. ML003740308, September 1976.

NOAA Hydrometeorological Report No. 52, "Application of Probable Maximum Precipitation Estimates – United States East of the 105th Meridian," U.S. Department of Commerce, National Oceanic and Atmospheric Administration, U.S. Department of the Army, Corps of Engineers, Washington, DC, August 1982.

NOAA Hydrometeorological Report No. 51, "Probable Maximum Precipitation Estimates - United States East of the 105th Meridian," U.S. Department of Commerce, National Oceanic and Atmospheric Administration, U.S. Department of the Army, Corps of Engineers, Washington, DC, 1978.

NOAA Hydrometeorological Report No. 53, "Seasonal Variation of 10-square mile Probable Maximum Precipitation Estimates – United States East of the 105th Meridian," U.S. Department of Commerce, National Oceanic and Atmospheric Administration, U.S. Department of the Army, Corps of Engineers, Washington, DC, 1980.

ANS 2.8-1992, "Determining Design Basis Flooding at Power Reactor Sites," 1992.

RECOMMENDATION 2.3: SEISMIC

**PURPOSE**

The U.S. Nuclear Regulatory Commission (NRC or Commission) is issuing this information request for the following purposes:

- To gather information with respect to Near-Term Task Force (NTTF) Recommendation 2.3, as amended by staff requirements memorandum (SRM) associated with SECY-11-0124 and SECY-11-0137,
- To request licensees to develop a methodology and acceptance criteria for seismic walkdowns to be endorsed by the NRC staff,
- To request licensees to perform seismic walkdowns using the NRC-endorsed walkdown methodology, as defined herein,
- To identify and address degraded, nonconforming, or unanalyzed conditions through the corrective action program, and
- To verify the adequacy of licensee monitoring and maintenance procedures.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(f), addressees are required to submit a written response to this information request.

**BACKGROUND**

Structures, systems, and components (SSCs) important to safety in operating nuclear power plants are designed either in accordance with, or meet the intent of, Appendix A to 10 CFR Part 100 and Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 2. GDC 2 states that SSCs important to safety at nuclear power plants must be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their intended safety functions. The design bases for these SSCs are to reflect appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area. The design bases are also to reflect sufficient margin to account for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

In response to the accident at the Fukushima Dai-ichi nuclear power plant caused by the March 11, 2011, Tohoku earthquake and subsequent tsunami, the Commission established the NTTF to conduct a systematic review of NRC processes and regulations and to make recommendations to the Commission for its policy direction. The NTTF developed a set of recommendations that are intended to clarify and strengthen the regulatory framework for protection against natural phenomena. The purpose of this letter is to gather information with respect to NTTF Recommendation 2.3 for seismic hazards. Recommendation 2.3, and the SRMs associated with SECY-11-0124 and SECY-11-0137, instructs the NRC staff to issue requests for information to licensees pursuant to 10 CFR 50.54(f). This information request is for licensees to develop a methodology and acceptance criteria for seismic walkdowns to be endorsed by the staff following interaction with external stakeholders. It is requested that licensees perform the seismic walkdowns to identify and address plant-specific vulnerabilities (through its corrective action program) and verify the adequacies of monitoring and maintenance procedures.

In developing Recommendation 2.3, the NTTF recognized the need to verify the adequacy of features that play an integral role in the defense-in-depth approach for protection from natural phenomena. NTTF Recommendation 2.3 and SECY-11-0124 and SECY-11-0137 states that recent plant inspections have been conducted by NRC staff and industry in response to the Fukushima Dai-ichi accident and that these activities should be used to inform the implementation of this recommendation. Ongoing inspections of the Fukushima Dai-ichi and Dai-ni nuclear power stations may also provide insights useful for this recommendation. Furthermore, recent lessons learned from the earthquake near the North Anna Power Station should also be used to inform the development of the walkdown procedure(s).

**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 50.34(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 meet the intent of, GDC 2 and 10 CFR Part 100, Appendix A. Appendix A requires that safety-related SSCs remain functional if the safe shutdown earthquake (SSE) occurs.

**DISCUSSION**

The NTTF recommended that the Commission direct several actions to ensure adequate protection from natural phenomena. The actions should be taken to prevent fuel damage, ensure containment integrity and the functionality of SSCs that support the spent fuel pool (SFP). In particular, NTTF Recommendation 2.3 states that the Commission should "Order licensees to perform seismic and flood protection walkdowns to identify and address plant-specific vulnerabilities and verify the adequacy of monitoring and maintenance for protection features such as water tight barriers and seals in the interim period until longer term actions are completed to update the design basis for external events." However, in the context of this letter, the NRC staff is focusing on degraded, nonconforming, or unanalyzed conditions.

The NRC staff's assessment of NTTF Recommendation 2.3 is discussed in SECY-11-0124. The NRC staff agreed with the NTTF Recommendation 2.3 findings and noted that various walkdown guidance exists and that recent plant inspections by staff in accordance with Temporary Instruction (TI) 2515/183, "Follow-up to the Fukushima Dai-ichi Nuclear Station Fuel Damage Event," and licensees' plant inspections in response to the Fukushima Dai-ichi accidents should help inform the implementation of this recommendation. Results of the NRC staff's evaluation of the recent earthquake near North Anna Power Station may also provide insights.

In its SRM to SECY-0124, the Commission approved the staff's proposed actions to implement without delay the NTTF recommendations as described in the SECY paper. With regard to Recommendation 2.3, the NRC staff-approved actions are to develop and issue a request for information to licensees pursuant to 10 CFR 50.54(f) to develop a methodology and acceptance criteria for seismic walkdowns to be endorsed by the NRC staff following interactions with external stakeholders, perform seismic walkdowns to identify and address plant-specific degraded, nonconforming, or unanalyzed conditions (through the corrective action program) and verify the adequacy of monitoring and maintenance for protective features, and inform the NRC staff of the results of the walkdowns and corrective actions taken or planned.

The TI 2515/183 was issued by the NRC on March 23, 2011. Inspection activities were completed by April 29, 2011, and NRC inspection reports were issued by May 13, 2011. The NRC developed a Summary of Observations report to encapsulate the performance of TI 2515/183 (see http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/follow-up-rpts.html). The summary report states that while individually, none of the observations posed a significant safety issue, they indicate a potential industry trend of failure to maintain equipment and strategies required to mitigate some design basis events. Regarding the licensees' capability to mitigate large fires or flooding coincident with seismic activity, the report notes that some equipment used to mitigate fires or station blackout was stored in areas that were not seismically qualified or that could be flooded.

As outlined in the SECY-11-0124, the NRC staff intends to work with the industry and other stakeholders to endorse a procedure(s) to develop acceptance criteria, conduct walkdowns, and identify degraded, nonconforming, or unanalyzed conditions. It is anticipated that the walkdown procedure will be developed by modifying various existing NRC and industry processes, including the recent inspections described above in accordance with TI 2515/183. Other guidance for seismic protection walkdowns include Electric Power Research Institute (EPRI) report NP-6041-SL, Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Seismic Qualification Utility Group procedure, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment," and International Atomic Energy Agency NS-G-2.13, "Evaluation of Seismic Safety for Existing Nuclear Installations." Additional details of attributes of a walkdown procedure are described in the Requested Action below.

The technical approach and methods used to develop the requested information should be integrated such that it accounts for design, physical barriers, procedures, temporary measures, and planned or installed mitigation measures to deal with external hazards. This type of an integrated approach will allow the NRC and industry to assess the significance of any new information related to the hazard in a systematic manner.

**REQUESTED ACTIONS**

In response to NTTF Recommendation 2.3, the Commission requests all licensees to perform seismic walkdowns in order to identify and address plant specific degraded, nonconforming, or unanalyzed conditions and verify the adequacy of strategies, monitoring, and maintenance programs such that the nuclear power plant can respond to external events. The walkdown will verify current plant configuration with the current licensing basis, verify the adequacy of current strategies, maintenance plans, and identify degraded, nonconforming, or unanalyzed conditions.

The walkdown procedure should be developed and submitted to the NRC. The procedure may incorporate current plant procedures, if appropriate. Prior to the walkdown, licensees should develop acceptance criteria, collect appropriate data, and assemble a team with relevant technical skills. Improvements made as part of the licensees' response to the individual plant examination of external events (IPEEE) program for seismic issues should be reported.

If any condition identified during the walkdown activities represents a degraded, nonconforming, or unanalyzed condition (i.e., noncompliance with the current licensing basis) for an SSC, describe actions that were taken or are planned to address the condition using the guidance in Regulatory Issues Summary 2005-20, Revision 1, Revision to NRC Inspection Manual Part 9900 Technical Guidance, "Operability Conditions Adverse to Quality or Safety," including entering the condition in the corrective action program. Reporting requirements pursuant to 10 CFR 50.72 should also be considered. Additionally, these findings should be considered in the Recommendation 2.1 hazard evaluations, as appropriate.

**REQUESTED INFORMATION**

1. The NRC requests that each addressee confirm that they will use the industry-developed, NRC-endorsed, seismic walkdown procedures[1] or provide a description of plant-specific walkdown procedures that include the following characteristics:

    a. Determination of the seismic walkdown scope and any combined effects
    b. Consideration of NUREG-1742, EPRI Report NP-6041, GIP, and common issues and findings discussed in the responses to TI 2515/183
    c. Pre-walkdown actions (e.g., data collection, review of drawings and procedures, identification of the plant licensing basis, identification of current seismic protection levels)
    d. Identification of SSCs requiring seismic protection and used in the protection of the reactor and spent fuel pool, including the ultimate heat sink (UHS)
    e. Description of the walkdown team composition and qualifications
    f. Details of the information to be collected during the walkdown including equipment access considerations
    g. Documentation and peer review requirements

2. Following the NRC's endorsement of the walkdown procedure, addresses are requested to conduct the walkdown and submit the final report which includes the following:

    a. Information on the plant-specific hazard licensing bases and a description of the protection and mitigation features considered in the licensing basis evaluation
    b. Information related to the implementation of the walkdown process
    c. A list of plant-specific 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)
    d. Results of the walkdown including key findings and identified degraded, nonconforming, or unanalyzed conditions. Include a detailed description of the

[1] NRC staff are currently engaged with industry and other external stakeholders to develop NRC-endorsed procedures. The NRC staff anticipates completing this activity by May, 2012.

actions taken or planned to address these conditions using the guidance in Regulatory Issues Summary 2005-20, Revision, 1, Revision to NRC Inspection Manual Part 9900 Technical Guidance, "Operability Conditions Adverse to Quality or Safety," including entering the condition in the corrective action program

e. Any planned or newly installed protection and mitigation features
f. Results and any subsequent actions taken in response to the peer review

**REQUIRED RESPONSE**

In accordance with 10 CFR 50.54(f), an addressee must respond as described below. The submission of the requested information is in stages to allow adequate time for further interactions with the stakeholders to provide clarifications, to develop implementation procedures and processes, and to develop the associated guidance as needed.

1. Within 120 days of the date of this information request, the addressee will confirm that they intend to use the NRC-endorsed seismic walkdown procedures, or provide to the NRC a description of the process that will be used to conduct the walkdowns and to develop the needed information.

2. Within 180 days of the NRC's endorsement of the walkdown process, each addressee will submit its final response. This response should include a list of any areas that are unable to be inspected due to inaccessibility and a schedule for when the walkdown will be completed.

If an addressee cannot meet the requested response date, the addressee must provide a response within 90 days of the date of this information request 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.

**Enclosure 3 Reference List**

SECY 11-0124, "Recommended Actions to be taken without Delay from the Near-Term Task Force Report," Agencywide Documents Access and Management System (ADAMS) Accession No. ML11245A158, September 9, 2011.

SECY 11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," ADAMS Accession No. ML11272A111, October 3, 2011.

SRM SECY 11-0124, "Recommended Action to be taken without Delay from the Near-Term Task Force Report," ADAMS Accession No. ML112911571, dated October 18, 2011.

SRM SECY 11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," ADAMS Accession No. ML113490055, dated December 15, 2011.

10 CFR 50.54 – "Conditions of Licenses"

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 50.34(a)(1), (a)(3), (a)(4), (b)(1), (b)(2), and (b)(4)

10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors"

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

Temporary Instruction 2515/183, "Follow-up to the Fukushima Dai-ichi Nuclear Station Fuel Damage Event"

Summary of Observations report to encapsulate the performance of TI 2515/183 (http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/follow-up-rpts.html).

Electric Power Research Institute (EPRI) report NP-6041-SL Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," August 1991.

Seismic Qualification Utility Group (SQUG) procedure: "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment,"

International Atomic Energy Agency (IAEA) NS-G-2.13, "Evaluation of Seismic Safety for Existing Nuclear Installations."

RECOMMENDATION 2.3: FLOODING

## PURPOSE

The U.S. Nuclear Regulatory Commission (NRC or Commission) is issuing this information request for the following purposes:

- To gather information with respect to Near-Term Task Force (NTTF) Recommendation 2.3, as amended by staff requirements memorandum (SRM) associated with SECY-11-0124 and SECY-11-0137,
- To request licensees to develop a methodology and acceptance criteria for flooding walkdowns to be endorsed by the NRC staff,
- To request licensees to perform flooding walkdowns using an NRC-endorsed walkdown methodology, as defined herein
- To identify and address degraded, nonconforming, or unanalyzed conditions through the corrective action program
- To identify and address cliff-edge effects through the corrective action program
- To verify the adequacy of licensee monitoring and maintenance procedures.

Pursuant to Title 10 of the *Code of Federal* Regulations (10 CFR), Section 50.54(f), addressees are required to submit a written response to this information request.

## BACKGROUND

Structures, systems, and components (SSCs) important to safety in operating nuclear power plants are designed either in accordance with, or meet the intent of, Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 2. GDC 2 states that SSCs important to safety at nuclear power plants must be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their intended safety functions. The design bases for these SSCs are to reflect appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area. The design bases are also to reflect sufficient margin to account for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

In response to the accident at the Fukushima Dai-ichi nuclear power plant caused by the March 11, 2011, Tohoku earthquake and subsequent tsunami, the Commission established the NTTF to conduct a systematic review of NRC processes and regulations, and to make recommendations to the Commission for its policy direction. The NTTF developed a set of recommendations that are intended to clarify and strengthen the regulatory framework for protection against natural phenomena. The purpose of this letter is to gather information related to NTTF Recommendation 2.3 for flooding hazards. Recommendations 2.3, and the SRMs associated with SECY-11-0124 and SECY-11-0137, instructs the NRC staff to issue requests for information to licensees pursuant to 10 CFR 50.54(f). This information request is for licensees to develop a methodology and acceptance criteria for flooding walkdowns to be endorsed by the NRC staff following interaction with external stakeholders. Licensees are requested to perform flood protection walkdowns to identify and address plant-specific

degraded, nonconforming, or unanalyzed conditions and cliff-edge effects (through the corrective action program) and verify the adequacy of monitoring and maintenance procedures.

In developing Recommendation 2.3, the NTTF observed that, "some plants have an overreliance on operator actions and temporary flood mitigation measures such as sandbagging, temporary flood walls and barriers, and portable equipment to perform safety functions." The NTTF report also states that, "the Task Force has concluded that flooding risks are of concern due to a 'cliff-edge' effect, in that the safely consequences of a flooding event may increase sharply with a small increase in the flooding level. Therefore, it would be very beneficial to safety for all licensees to confirm that SSCs important to safely are adequately protected from floods."

The NRC, in the past, has developed regulatory programs aimed at identifying plant-specific vulnerabilities to external flooding hazards. In June of 1991, the NRC issued Supplement 4 to Generic Letter (GL) 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f)." This GL requested that "each licensee perform an individual plant examination of external events to identify vulnerabilities, if any, to severe accidents and report the results together with any licensee determined improvements and corrective actions to the Commission." Flood-related hazards were considered in the IPEEE program as one of the high winds, floods, and other (HFO) external initiating-event hazards. Of the 70 IPEEE submittals, most indicated some type of walkdown was performed for the HFO events. However, NUREG-1742 states, "the [HFO walkdown] submittals usually did not provide detailed descriptions of the walkdown procedures and results." NUREG-1742 also states that, "A few licensees proposed flood-related countermeasures that may be optimistic. For example, one licensee took credit for sandbagging up to a level of 9 feet. In several other submittals, flood barriers made of various construction materials, such as logs or concrete berms, were credited with being effective for preventing flooding, but the submittals did not discuss whether the licensees performed confirmatory testing to verify the effectiveness of certain of these mitigating actions."

In late December 1999, a severe storm induced flooding at Le Blayais nuclear power plant site in France. Lessons learned from this flooding event are documented in World Association of Nuclear Operators Significant Event Report (SER) 2000-3, "Severe Storm Results in Scram of Three Units and Loss of Safety System Functions due to Partial Plant Flooding," and in Institute of Nuclear Power Operations (INPO) SER 1-01, with the same title. Both reports list significant aspects and important lessons learned from the flooding event. On March 11, 2010, Électricité de France presented lessons learned from the 1999 Blayais flood at the NRC's Regulatory Information Conference (http://www.nrc.gov/public-involve/conference-symposia/ric/past/2010/slides/th35defraguierepv.pdf). Lessons learned discussed in this presentation were: (1) cable openings and trenches were an unrecognized common-mode vulnerability requiring review of existing protective measures, (2) difficulty in detecting water in affected rooms and an inadequate warning system, and (3) the flood's effects on support functions and surrounding areas were not adequately accounted or were inappropriate for the weather conditions.

**APPLICABLE REGULATORY REQUIREMENTS**

- 10 CFR 50.34(a)(1), (a)(3), (a)(4), (b)(1), (b)(2), and (b)(4)
- 10 CFR 50.54, "Conditions of Licenses"
- Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, GDC 2, "Design Bases for Protection against Natural Phenomena"
- Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100

The flooding design bases for currently operating nuclear power plants were either developed in accordance with, or meet the intent of, GDC 2 and 10 CFR Part 100, Appendix A (seismically induced floods and water waves). GDC 2 states that SSCs important to safety at nuclear power plants must be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their intended safety functions. The design bases for these SSCs are to reflect appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area. The design bases are also to reflect sufficient margin to account for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

**DISCUSSION**

The NTTF recommended that the Commission direct several actions to ensure adequate protection from natural phenomena. These actions should be taken to prevent fuel damage and to ensure containment and spent fuel pool integrity. In particular, Recommendation 2.3 states that the Commission should "Order licensees to perform seismic and flood protection walkdowns to identify and address plant-specific vulnerabilities and verify the adequacy of monitoring and maintenance for protection features such as water tight barriers and seals in the interim period until longer term actions are completed to update the design basis for external events." However, in the context of this letter, the NRC staff is focusing on degraded, nonconforming, or unanalyzed conditions and cliff-edge effects.

The NRC staff's assessment of NTTF Recommendation 2.3 is discussed in SECY-11-0124. The NRC staff agreed with the NTTF Recommendation 2.3 findings and noted that some plants rely on operator actions and temporary flood mitigation measures such as sandbagging, temporary flood walls and barriers, and portable equipment to perform safety functions. Results of staff's inspections at nuclear power sites in accordance with Temporary Instruction (TI) 2515/183 identified potential issues and observations regarding mitigation measures. Recent flooding at the Fort Calhoun site showed the importance of temporary flood mitigation measures. The NRC staff also noted that guidance should be developed for flooding walkdowns with external stakeholder involvement to ensure consistency.

In its SRM to SECY-11-0124, the Commission approved the NRC staff's proposed actions to implement without delay the NTTF recommendations as described in the SECY. With regards to Recommendation 2.3, NRC staff's approved actions are to develop and issue a request for information to licensees pursuant to 10 CFR 50.54(f) to develop a methodology and acceptance criteria for flooding walkdowns to be endorsed by the NRC staff following interaction with external stakeholders, perform flood protection walkdowns to identify and address plant-specific degraded, nonconforming, or unanalyzed conditions and cliff-edge effects (through the corrective action program) and verify the adequacy of monitoring and maintenance for protection features, and inform the NRC of the results of the walkdowns and corrective actions taken or planned.

The TI 2515/183 was issued by the NRC on March 23, 2011. Inspection activities were completed by April 29, 2011, and NRC inspection reports were issued by May 13, 2011. The NRC developed a Summary of Observations report to document the performance of TI 2515/183 (see http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/follow-up-rpts.html). The summary report states that while individually, none of the observations posed a significant safety issue, they indicate a potential industry trend of failure to maintain equipment and strategies required to mitigate some design basis events. Regarding the licensee's capability to mitigate design bases flooding events, the report notes that some equipment (mainly pumps) would not operate when tested, or lacked test acceptance criteria, and that some discrepancies were identified with barrier and penetration seals.

Additional review of Section 03.03 of the responses to TI 2515/183 indicates that several sites were susceptible to water accumulation that submerged safety-related cables. Issues were noted with cracks in penetrations, evidence of water infiltration, and groundwater intrusion. Individual TI inspection reports noted that a few licensee-proposed flood-related countermeasures may not achieve the intended mitigative effect. Flood barriers made of various construction materials were credited with being effective for preventing flooding, but the confirmatory testing to verify the effectiveness of certain of these mitigating actions was not conclusive. It should be noted that these findings are consistent with findings documented in the "Perspectives Gained" section of the IPEEE program report (NUREG-1742).

The Advisory Committee on Reactor Safeguards (ACRS) in its letter dated October 13, 2011, requested that the Commission consider that "site-specific external hazards, vulnerabilities, and consequences need to be evaluated in an integrated context. For example, tornadoes and hurricanes may cause extended loss of offsite power with coincident physical damage to nonsafety structures or equipment at multiple units that has not been fully evaluated. Damage from severe storms or other site-specific hazards may also disable external essential cooling water supplies. Vulnerabilities to those hazards and subsequent damage may not be identified from assessments that focus only on design-basis seismic and flooding events." The ACRS further requested that "Near-term actions related to NTTF Recommendation 2.3 should be expanded to assure that the walkdowns address the integrated effects of severe storms as well as seismic and flooding events. The walkdowns and associated assessments should confirm that the identified hazards and vulnerabilities remain bounded by the current plant licensing basis."

The NRC staff will interact with industry and stakeholders to develop a methodology and acceptance criteria for flooding walkdowns. These walkdowns should integrate the External Flood results in NUREG-1742, common issues and findings discussed in Section 03.03 of the responses to TI 2515/183, and the Significant Aspect findings discussed INPO SER 1-01. It is anticipated that the walkdown procedure will be developed or modified using various existing NRC- and industry-developed procedures. As mentioned in SECY-11-0124, recent flood events such as those at Fort Calhoun should also provide valuable insights. Additional attributes of the walkdown procedure are described in the Requested Action section below. The technical approach used to develop the needed information should be holistic and integrated to account for the site-specific design, physical barriers, procedures, temporary measures, and planned or installed mitigation measures to deal with the potential flooding scenarios.

As stated earlier, the NRC staff will interact with industry and other stakeholders to develop an approach, which can be applied in a uniform and consistent manner across the different sites and plant conditions. An integrated approach will allow the NRC and industry to assess the significance of any new information related to flooding hazards in a systematic manner. During these interactions, the NRC staff will also work with industry and stakeholders to identify efficiencies and strategies to ensure that responses and reviews are timely and support the Commission guidance on the overall schedule.

As mentioned in the cover letter, other external events (e.g., extreme winds and its effects) will be covered as a separate action from this letter. It would be prudent for addressees to consider the inclusion of other external events in these walkdown procedures due to the potential efficient use of similar resources to perform these walkdowns.

**REQUESTED ACTIONS**

The NRC requests that each addressee confirm that they will use the industry-developed, NRC-endorsed, flood walkdown procedures[1] or provide a description of plant-specific walkdown procedures. The requested actions include the following:

(1) Perform flood protection walkdowns using an NRC-endorsed walkdown methodology,
(2) Identify and address plant-specific degraded, nonconforming, or unanalyzed conditions, as well as, cliff-edge effects through the corrective action program, and consider these findings in the Recommendation 2.1 hazard evaluations, as appropriate,
(3) Identify any other actions taken or planned to further enhance the site flood protection,
(4) Verify the adequacy of programs, monitoring and maintenance for protection features, and,
(5) Report to the NRC the results of the walkdowns and corrective actions taken or planned.

A final report should be submitted to the NRC addressing items identified in the Requested Information section.

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[1] NRC staff are currently engaged with industry and other external stakeholders to develop NRC-endorsed procedures. The NRC staff anticipates completing this activity by May, 2012.

It is requested that the walkdown procedure verify that flood protection systems for the plant are available, functional, and implementable under a variety of site conditions. In particular, the walkdowns should confirm that: (1) cable and piping trenches and other penetrations to SSCs important to safety, including underground rooms, are not pathways for external ingress of water, (2) adequate water detection and warning systems are available, if credited in the current licensing basis, (3) the effects of elevated water levels and severe weather conditions would not impair support functions or would not impede performing necessary actions given the weather conditions, and (4) other factors at multi-unit sites (e.g., equipment availability and staffing) would not prevent implementation of flood protection measures.

If any condition identified during the walkdown activities represents a degraded, nonconforming, or unanalyzed condition (i.e., noncompliance with the current licensing basis) for an SSC, describe actions that were taken or are planned to address the condition using the guidance in Regulatory Issues Summary 2005-20, Revisions 1, Revision to NRC Inspection Manual Part 9900 Technical Guidance, "Operability Conditions Adverse to Quality or Safety," including entering the condition in the corrective action program. Reporting requirements pursuant to 10 CFR 50.72 should also be considered. In addition, if any condition noted during the walkdown represents a cliff-edge effect, describe any measures taken or planned to address the condition(s) while the corrective action is being implemented.

Along with an assessment of reactor integrity, the NTTF recommended an evaluation of spent fuel pools to assess the effectiveness of the flood protection. The approach should account for the site-specific design, physical barriers, procedures, temporary measures, and planned or existing mitigation measures.

**REQUESTED INFORMATION**

1. The NRC requests that each addressee confirm that it will use the industry-developed, NRC-endorsed, flooding walkdown procedures or provide a description of plant-specific walkdown procedures that include the following characteristics:

   a. Address the NTTF Report's observations regarding "overreliance on operator actions and temporary flood mitigation measures" and the 'cliff-edge' effect regarding a sharp increase in flooding risks with a small increase in flooding level.
   b. Integrate issues discussed in the External Flood Qualitative Results (Section 4.3.3) in NUREG-1742, common issues and findings discussed in Section 03.03 of the responses to TI 2515/183, and the Significant Aspect findings discussed in INPO SER 1-01.
   c. Integrate insights from any new and relevant flood hazard information, as well as recent flood-related walkdowns such as the events at the Fort Calhoun site, as mentioned in SECY-11-0124. Additionally, relevant NRC inspection findings could provide additional insights.
   d. Integrate the combined effects of flooding along with other adverse conditions, such as high winds, hail, lightning, etc., that could reasonably be expected to simultaneously occur. For example, steps in a flooding procedure that require manipulation of systems and components in outside areas of the plant site that could not be safely assessed because of storm conditions.

e. Identify pre-walkdown actions, such as the collection of current site topography including any changes since the original licensing (e.g., security improvements and temporary structures), sets of as-built drawings, review of the existing design basis flood level(s), review of any flood protection and pertinent flood mitigation features, such as exterior barriers, incorporated barriers, and temporary flood barriers.
f. Identify a list of pertinent elevations of Regulatory Guide 1.29[2] structures, systems, and components that should be designed to withstand the design basis hazard (similar to Table 1, i.e., 3.1.3 of American National Standards Institute/American Nuclear Society (ANSI/ANS)-2.8-1992)
g. Identify the team composition and qualifications.
h. Verify that flood protection systems are available, functional, and implementable under a variety of site conditions by reviewing the following:
    i. Operator availability, operator training, timeliness of response, equipment maintenance and operability, back-up availability, operator access under adverse site conditions[3]
    ii. Methods and acceptance criteria to evaluate exterior barriers[4]
    iii. Methods and acceptance criteria to evaluate incorporated barriers
    iv. Methods and acceptance criteria to evaluate temporary flood barriers
    v. Preparations in advance of adverse weather conditions
i. Identify programs in place that periodically verify the status and adequacy of flood mitigation strategies and equipment.
j. Develop a documentation template, including peer-review requirements, so that walkdown results can be efficiently and uniformly reviewed and evaluated. The template should also consider the reporting requirement discussed below.

2. Following NRC's endorsement of the walkdown procedure, conduct the walkdown and submit a final report which includes the following:

    a. Describe the design basis flood hazard level(s) for all flood-causing mechanisms, including groundwater ingress.
    b. Describe protection and mitigation features that are considered in the licensing basis evaluation to protect against external ingress of water into SSCs important to safety.
    c. Describe any warning systems to detect the presence of water in rooms important to safety.
    d. Discuss the effectiveness of flood protection systems and exterior, incorporated, and temporary flood barriers. Discuss how these systems and barriers were evaluated using the acceptance criteria developed as part of Requested Information item 1.h.

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[2] Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants", and Regulatory Guide 1.102, Flood Protection for Nuclear Power Plants," both recommend the use of Regulatory Guide 1.29, "Seismic Design Classification" for identifying structures, systems, and components, that should be designed to withstand the conditions resulting from the design basis flood and remain functional.
[3] This may not be an all-inclusive list.
[4] See Regulatory Position 1 of Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," for definitions acceptable to the NRC staff for exterior barriers, incorporated barriers, and temporary barriers.

e. Present information related to the implementation of the walkdown process (e.g., details of selection of the walkdown team and procedures,) using the documentation template discussed in Requested Information item 1.j, including actions taken in response to the peer review.
f. Results of the walkdown including key findings and identified degraded, nonconforming, or unanalyzed conditions. Include a detailed description of the actions taken or planned to address these conditions using the guidance in Regulatory Issues Summary 2005-20, Revision 1, Revision to NRC Inspection Manual Part 9900 Technical Guidance, "Operability Conditions Adverse to Quality or Safety," including entering the condition in the corrective action program.
g. Document any cliff-edge effects identified and the associated basis. Indicate those that were entered into the corrective action program. Also include a detailed description of the actions taken or planned to address these effects.
h. Describe any other planned or newly installed flood protection systems or flood mitigation measures including flood barriers that further enhance the flood protection. Identify results and any subsequent actions taken in response to the peer review.

**REQUIRED RESPONSE**

In accordance with 10 CFR 50.54(f), an addressee must respond as described below. The submission of the requested information is in stages to allow adequate time for further interactions with the stakeholders to provide clarifications, to develop implementation procedures and processes, and to develop the associated guidance as needed.

1. Within 90 days of the date of this information request, the addressee will confirm that it intends to use the NRC-endorsed flooding walkdown procedures or provide the NRC a description of the process that will be used to conduct the walkdowns and to develop the needed information.

2. Within 180 days of NRC's endorsement of the walkdown procedure, each addressee will submit its final response for the requested information. This response should include a list of any areas that are unable to be inspected due to inaccessibility and a schedule for when the walkdown will be completed.

If an addressee cannot meet the requested response date, the addressee must provide a response within 90 days of the date of this information request 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.

**Enclosure 4 References**

SECY-11-0124, "Recommended Actions to be taken without Delay from the Near-Term Task Force Report," Agencywide Documents Access and Management System (ADAMS) Accession No. ML11245A158, dated September 9, 2011.

SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons-Learned," ADAMS Accession No. ML11272A111, October 3, 2011.

"Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Dai-ichi Accident," ADAMS Accession No. ML111861807, July 12, 2011.

10 CFR 50.54 – Conditions of Licenses

10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors"

Appendix A to 10 CFR Part 50, General Design Criteria for Nuclear Power Plants

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

Temporary Instruction 2515/183, "Follow-up to the Fukushima Dai-ichi Fuel Damage Event," November 2011, ADAMS Accession No. ML113220407.

Energy and Water Development and Related Agencies Appropriations Act, 2012

NUREG-0800, SRP Section 2.4

NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Final Report, ADAMS Accession No. ML063550238, June 1991.

ASME/ANS RA-Sa-2009, 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," 2009.

INPO version, SER 1–01, "WANO Significant Event Report (SER) 2000-3, 'Severe Storm Results in Scram of Three Units and Loss of Safety System Functions Due to Partial Plant Flooding,'" February 2001 (Proprietary)

## RECOMMENDATION 9.3: EMERGENCY PREPAREDNESS

### Communications

### PURPOSE

The U.S. Nuclear Regulatory Commission (NRC or Commission) is issuing this information request regarding the power supplies for communications systems to determine if additional regulatory action is warranted. This request is based upon Near-Term Task Force (NTTF) Recommendation 9.3 which proposed that facility emergency plans provide for a means to power communications equipment needed to communicate onsite (e.g., radios for response teams and between facilities) and offsite (e.g., cellular telephones and satellite telephones) during a prolonged station blackout.

### APPLICABLE REGULATORY REQUIREMENTS AND GUIDANCE

Emergency plan communications requirements and detailed guidance on how to meet those requirements are contained in the following:

1. Title 10 of the Code *of Federal Regulations* (10 CFR) 50.47 (b)(6) states that provisions should be made for prompt communications among principal response organizations to emergency personnel and to the public.

2. Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, "Domestic Licensing for Production and Utilization Facilities," Section IV. E. 9, states that adequate provisions shall be made and described for emergency facilities and equipment, including "at least one onsite and one offsite communications system; each system shall have a backup power source."

3. NUREG-0696, "Functional Criteria for Emergency Response Facilities," issued February 1981, offers guidance on how to meet the requirements of Appendix E to 10 CFR Part 50, and discusses the onsite and offsite communications requirements for the licensee's emergency operating facilities.

### DISCUSSION

During the March 11, 2011, Tōhoku earthquake and subsequent tsunami, the widespread destruction and loss of electrical power degraded communications capabilities onsite at Fukushima Dai-ichi and between the site and external stakeholders, such as local emergency response centers, the Japanese government, and corporate offices. Normal and emergency offsite communications systems lost power or were degraded by the earthquake and tsunami. Normal and emergency onsite communications were severely impacted by the loss of power to signal repeaters and depleted radio batteries. Accounts of the accident response refer to delays in repair activities caused by issues with the ability to effectively communicate between repair teams and the control rooms and the onsite emergency response center.

The NRC requests that the following assumptions be made in preparing responses to this request for information: the potential onsite and offsite damage is a result of a large scale natural event resulting in a loss of all alternating current (ac) power.

In addition, assume that the large scale natural event causes extensive damage to normal and emergency communications systems both onsite and in the area surrounding the site. It has been recognized that following a large scale natural event that ac power may not be available to cell and other communications infrastructures.

**REQUESTED ACTIONS**

It is requested that addressees assess their current communications systems and equipment used during an emergency event given the aforementioned assumptions. It is also requested that consideration be given to any enhancements that may be appropriate for the emergency plan with respect to communications requirements of 10 CFR 50.47, Appendix E to 10 CFR Part 50, and the guidance in NUREG-0696 in light of the assumptions stated above. Also addressees are requested to consider the means necessary to power the new and existing communications equipment during a prolonged SBO.

**REQUESTED INFORMATION**

1. Addressees are requested to provide an assessment of the current communications systems and equipment used during an emergency event to identify any enhancements that may be needed to ensure communications are maintained during a large scale natural event meeting the conditions described above. The assessment should:

    - Identify any planned or potential improvements to existing onsite communications systems and their required normal and/or backup power supplies,

    - Identify any planned or potential improvements to existing offsite communications systems and their required normal and/or backup power supplies,

    - Provide a description of any new communications system(s) or technologies that will be deployed based upon the assumed conditions described above, and

    - Provide a description of how the new and/or improved systems and power supplies will be able to provide for communications during a loss of all ac power,

2. Addressees are requested to describe any interim actions that have been taken or are planned to be taken to enhance existing communications systems power supplies until the communications assessment and the resulting actions are complete,

3. Provide an implementation schedule of the time needed to conduct and implement the results of the communications assessment.

**REQUIRED RESPONSE**

The addressee should respond to this request for information no later than 90 days from the date of issuance.

If an addressee cannot meet the requested response date, the addressee must provide a response within 60 days of the date of this letter 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 date.

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.

## Staffing

### PURPOSE

The U.S. Nuclear Regulatory Commission (NRC or Commission) is issuing this information request to determine if additional regulatory action is warranted regarding the staff required to fill all necessary positions to respond to a multi-unit event.

Single unit sites should provide the requested information as it pertains to an extended loss of all ac power, and impeded access to the site.

### APPLICABLE REGULATORY REQUIREMENTS AND GUIDANCE

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.47(b)(1) states, in part: "... and each principal response organization has staff to respond and to augment its initial response on a continuous basis."

- 10 CFR 50.47(b)(2) states, in part: "... adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and..."

- NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Section B, Onsite Emergency Organization, states in part:

> Each licensee shall specify... functional areas of emergency activity... These assignments shall cover the emergency functions in Table B-1 entitled, 'Minimum Staffing Requirements for Nuclear Power Plant Emergencies.' The minimum on-shift staffing shall be as indicated in Table B-1. The licensee must be able to augment on-shift capabilities within a short period after declaration of an emergency. This capability shall be as indicated in Table B-1...

### DISCUSSION

The events in Japan have highlighted the importance of responders during all phases of emergency event response. The regulations require emergency response capabilities during a broad spectrum of postulated reactor accidents. A natural event on the scale of the 2011 Great East Japan Earthquake and resulting tsunami could present new challenges to personnel and their safety. Specifically, the event stressed the existing regulatory framework and impacted the operator's capability to implement adequate protective measures to protect the public and plant staff. In light of the experience from the event, the unavailability of sufficient onsite staff during the initial phase of the emergency condition, the unavailability of staff designated to augment the onsite staff, the inability for offsite support to reach the site, and the unavailability and inability of relief staff to reach the site, the NRC recognizes that these in total could pose challenges to licensee response efforts.

A large scale natural event may alter the planned emergency framework by changing access routes (e.g., bridges washed out, debris blocking roadways, etc.). While several utilities have implemented a combined emergency operations facility that is capable of handling multi-unit events, the onsite technical support center and operational support center at sites with multiple reactors have been designed to handle any emergency at only one of the units.

In conjunction with the Emergency Preparedness regulations Agencywide Documents Access and Management System (ADAMS) Accession No. ML112070125 published on November 10, 2011, the NRC published on December 5, 2011, in the *Federal Register* (76 FR 75771) interim staff guidance (ISG) in NSIR/DPR-ISG-01 ADAMS Accession No. ML113010523. Section IV.C of the ISG provides guidance on performing an on-shift staffing analysis, and identified Nuclear Energy Institute (NEI)-10-05, "Assessment of On-shift Emergency Response Organizations Staffing and Capabilities" ADAMS Accession No. ML111751698, as an acceptable methodology for such an analysis. However, this methodology and guidance does not consider multiple unit events involving a large scale natural event with a loss of all alternating current (ac) power.

This letter requests that addresses assess and provide the NRC with information regarding the ability to implement their emergency plan during a large scale natural event that results in the following:

- all units affected,
- extended loss of all ac power, and
- impeded access to the sites

Addressees may find the capability for assessment activities, including repair team planning and preparation are particularly impacted. Therefore, it is requested that this assessment ensure that there is sufficient onsite staff and other resources to perform critical tasks until augmentation staff arrives to provide assistance and until other offsite resources become available.

**REQUESTED ACTIONS**

It is requested that addressees assess their current staffing levels and determine the appropriate staff to fill all necessary positions for responding to a multi-unit event during a beyond design basis natural event and determine if any enhancements are appropriate given the considerations of Near-Term Task Force (NTTF) Recommendation 9.3.

Single unit sites should provide the requested information as it pertains to an extended loss of all ac power, and impeded access to the site.

**REQUESTED INFORMATION**

1. It is requested that addressees provide an assessment of the onsite and augmented staff needed to respond to a large scale natural event meeting the conditions described above. This assessment should include a discussion of the onsite and augmented staff available to implement the strategies as discussed in the emergency plan and/or described in plant operating procedures. The following functions are requested to be assessed:

- How onsite staff will move back-up equipment (e.g., pumps, generators) from alternate onsite storage facilities to repair locations at each reactor as described in the Order regarding the NTTF Recommendation 4.2. It is requested that consideration be given to the major functional areas of NUREG-0654, Table B-1, such as plant operations and assessment of operational aspects, emergency direction and control, notification/communication, radiological accident assessment, and support of operational accident assessment, as appropriate.

- New staff or functions identified as a result of the assessment.

- Collateral duties (personnel not being prevented from timely performance of their assigned functions).

2. Provide an implementation schedule of the time needed to conduct the onsite and augmented staffing assessment. If any modifications are determined to be appropriate, please include in the schedule the time to implement the changes.

3. Identify how the augmented staff would be notified given degraded communications capabilities.

4. Identify the methods of access (e.g., roadways, navigable bodies of water and dockage, airlift, etc.) to the site that are expected to be available after a widespread large scale natural event.

5. Identify any interim actions that have been taken or are planned prior to the completion of the staffing assessment.

6. Identify changes that have been made or will be made to your emergency plan regarding the on-shift or augmented staffing changes necessary to respond to a loss of all ac power, multi-unit event, including any new or revised agreements with offsite resource providers (e.g., staffing, equipment, transportation, etc.).

**REQUIRED RESPONSE**

In accordance with Section 182.a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), each addressee is requested to submit a written response consistent with the requested information. The response to requested information items 1 and 2 should be provided within 60 days of issuance of the ISG to be referenced in the NRC Order associated with NTTF Recommendation 4.2. The response to requested information items 3-6 should be provided within 90 days of the date of this letter.

If an addressee cannot meet the requested response date, the addressee must provide a response within 60 days of the date of this letter 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 date.

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.

POWER REACTOR LICENSEES AND HOLDERS OF
CONSTRUCTION PERMITS IN ACTIVE OR DEFERRED STATUS

<u>Arkansas Nuclear One</u>
Entergy Operations, Inc.
Docket Nos. 50-313 and 50-368
License Nos. DPR-51 and NPF-6

Mr. Christopher J. Schwarz
Vice President, Operations
Entergy Operations, Inc.
Arkansas Nuclear One
1448 S.R. 333
Russellville, AR 72802

<u>Beaver Valley Power Station</u>
First Energy Nuclear Operating Co.
Docket Nos. 50-334 and 50-412
License Nos. DPR-66 and NPF-73

Mr. Paul A. Harden
Site Vice President
FirstEnergy Nuclear Operating Company
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

<u>Bellefonte Nuclear Power Station</u>
Tennessee Valley Authority
Docket Nos. 50-438 and 50-439
Construction Permit Nos. CPPR No. 122 and CPPR No. 123

Mr. Michael D. Skaggs
Senior Vice President, Nuclear Generation Development and Construction
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

<u>Braidwood Station</u>
Exelon Generation Co., LLC
Docket Nos. STN 50-456 and STN 50-457
License Nos. NPF-72 and NPF-77

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

<u>Browns Ferry Nuclear Plant</u>
Tennessee Valley Authority
Docket Nos. 50-259, 50-260 and 50-296
License Nos. DPR-33, DPR-52 and DPR-68

Mr. Preston D. Swafford
Chief Nuclear Officer and Executive Vice President
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

<u>Brunswick Steam Electric Plant</u>
Carolina Power & Light Co.
Docket Nos. 50-325 and 50-324
License Nos. DPR-71 and DPR-62

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Vice President
Carolina Power & Light Company
Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461

<u>Byron Station</u>
Exelon Generation Co., LLC
Docket Nos. STN 50-454 and STN 50-455
License Nos. NPF-37 and NPF-66

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

<u>Callaway Plant</u>
Union Electric Co.
Docket No. 50-483
License No. NPF-30

Mr. Adam C. Heflin
Senior Vice President and Chief Nuclear Officer
Union Electric Company
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Fulton, MO 65251

Calvert Cliffs Nuclear Power Plant
Calvert Cliffs Nuclear Power Plant, LLC
Docket Nos. 50-317 and 50-318
License Nos. DPR-53 and DPR-69

Mr. George H. Gellrich
Vice President
Calvert Cliffs Nuclear Power Plant, LLC
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

Catawba Nuclear Station
Duke Energy Carolinas, LLC
Docket Nos. 50-413 and 50-414
License Nos. NPF-35 and NPF-52

Mr. James R. Morris
Site Vice President
Duke Energy Carolinas, LLC
Catawba Nuclear Station
4800 Concord Road
York, SC 29745

Clinton Power Station
Exelon Generation Co., LLC
Docket No. 50-461
License No. NPF-62

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Columbia Generating Station
Energy Northwest
Docket No. 50-397
License No. NPF-21

Mr. Mark E. Reddemann
Chief Executive Officer
Energy Northwest
MD 1023
P.O. Box 968
Richland, WA 99352

<u>Comanche Peak Nuclear Power Plant</u>
Luminant Generation Co., LLC
Docket Nos. 50-445 and 50-446
License Nos. NPF-87 and NPF-89

Mr. Rafael Flores
Senior Vice President and Chief Nuclear Officer
Luminant Generation Company, LLC
Attn: Regulatory Affairs
P. O. Box 1002
Glen Rose, TX 76043

<u>Cooper Nuclear Station</u>
Nebraska Public Power District
Docket No. 50-298
License No. DPR-46

Mr. Brian J. O'Grady
Vice President - Nuclear and Chief Nuclear Officer
Nebraska Public Power District
72676 648A Avenue
P.O. Box 98
Brownville, NE 68321

<u>Crystal River Nuclear Generating Plant</u>
Florida Power Corp.
Docket No. 50-302
License No. DPR-72

Mr. Jon A. Franke
Vice President
Attn: Supervisor, Licensing & Regulatory Affairs
Progress Energy, Inc.
Crystal River Nuclear Plant (NA2C)
15760 West Power Line Street
Crystal River, FL 34428-6708

<u>Davis-Besse Nuclear Power Station</u>
First Energy Nuclear Operating Co.
Docket No. 50-346
License No. NPF-3

Mr. Barry S. Allen
Site Vice President
FirstEnergy Nuclear Operating Company
c/o Davis-Besse NPS
5501 N. State Route 2
Oak Harbor, OH 43449-9760

<u>Diablo Canyon Power Plant</u>
Pacific Gas & Electric Co.
Docket Nos. 50-275 and 50-323
License Nos. DPR-80 and DPR-82

Mr. John T. Conway
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Pacific Gas and Electric Company
Diablo Canyon Power Plant
77 Beale Street, Mail Code B32
San Francisco, CA 94105

<u>Donald C. Cook Nuclear Plant</u>
Indiana Michigan Power Co.
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

Mr. Lawrence J. Weber
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Indiana Michigan Power Company
Nuclear Generation Group
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<u>Dresden Nuclear Power Station</u>
Exelon Generation Co., LLC
Docket Nos. 50-237 and 50-249
License Nos. DPR-19 and DPR-25

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

<u>Duane Arnold Energy Center</u>
NextEra Energy Duane Arnold, LLC
Docket No. 50-331
License No. DPR-49

Mr. Peter Wells
Site Vice President
NextEra Energy
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

Edwin I. Hatch Nuclear Plant
Southern Nuclear Operating Co.
Docket Nos. 50-321 and 50-366
License Nos. DPR-57 and NPF-5

Mr. Dennis R. Madison
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Edwin I. Hatch Nuclear Plant
11028 Hatch Parkway North
Baxley, GA 31513

Fermi
Detroit Edison Co.
Docket No. 50-341
License No. NPF-43

Mr. Jack M. Davis
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Detroit Edison Company
Fermi 2 – 210 NOC
6400 North Dixie Highway
Newport, MI 48166

Fort Calhoun Station
Omaha Public Power District
Docket No. 50-285
License No. DPR-40

Mr. David J. Bannister
Vice President and Chief Nuclear Officer
Omaha Public Power District
444 South 16th St. Mall
Omaha, NE 68102-2247

Grand Gulf Nuclear Station
Entergy Operations, Inc.
Docket No. 50-416
License No. NPF-29

Mr. Michael Perito
Vice President, Operations
Entergy Operations, Inc.
Grand Gulf Nuclear Station, Unit 1
7003 Bald Hill Road
Port Gibson, MS 39150

H. B. Robinson Steam Electric Plant
Carolina Power & Light Co.
Docket No. 50-261
License No. DPR-23

Mr. Robert J. Duncan II
Vice President
Carolina Power & Light Company
3581 West Entrance Road
Hartsville, SC 29550

Hope Creek Generating Station
PSEG Nuclear, LLC
Docket No. 50-354
License No. NPF-57

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
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Hancocks Bridge, NJ 08038

Indian Point Energy Center
Entergy Nuclear Operations, Inc.
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

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Indian Point Energy Center
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Buchanan, NY 10511-0249

James A. FitzPatrick Nuclear Power Plant
Entergy Nuclear Operations, Inc.
Docket No. 50-333
License No. DPR-59

Mike Colomb
Vice President, Operations
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Joseph M. Farley Nuclear Plant
Southern Nuclear Operating Co.
Docket Nos. 50-348 and 50-364
License Nos. NPF-2 and NPF-8

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Southern Nuclear Operating Company, Inc.
Joseph M. Farley Nuclear Plant
7388 North State Highway 95
Columbia, AL 36319

Kewaunee Power Station
Dominion Energy Kewaunee, Inc.
Docket No. 50-305
License No. DPR-43

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LaSalle County Station
Exelon Generation Co., LLC
Docket Nos. 50-373 and 50-374
License Nos. NPF-11 and NPF-18

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Limerick Generating Station
Exelon Generation Co., LLC
Docket Nos. 50-352 and 50-353
License Nos. NPF-39 and NPF-85

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<u>Millstone Nuclear Power Station</u>
Dominion Nuclear Connecticut, Inc.
Docket Nos. 50-336 and 50-423
License Nos. DPR-65 and NPF-49

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<u>Monticello Nuclear Generating Plant</u>
Northern States Power Company
Docket No. 50-263
License No. DPR-22

Mr. Timothy J. O'Connor
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Northern States Power Company - Minnesota
Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362-9637

<u>Nine Mile Point Nuclear Station</u>
Nine Mile Point Nuclear Station, LLC
Docket Nos. 50-220 and 50-410
License Nos. DPR-63 and NPF-69

Mr. Ken Langdon
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
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<u>North Anna Power Station</u>
Virginia Electric & Power Co.
Docket Nos. 50-338 and 50-339
License Nos. NPF-4 and NPF-7

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<u>Oconee Nuclear Station</u>
Duke Energy Carolinas, LLC
Docket Nos. 50-269, 50-270 and 50-287
License Nos. DPR-38, DPR-47 and DPR-55

Mr. Preston Gillespie
Site Vice President, Oconee Nuclear Station
Duke Energy Carolinas, LLC
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<u>Oyster Creek Nuclear Generating Station</u>
Exelon Generation Co., LLC
Docket No. 50-219
License No. DPR-16

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<u>Palisades Nuclear Plant</u>
Entergy Nuclear Operations, Inc.
Docket No. 50-255
License No. DPR-20

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Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
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<u>Palo Verde Nuclear Generating Station</u>
Arizona Public Service Company
Docket Nos. STN 50-528, STN 50-529 and STN 50-530
License Nos. NPF-41, NPF-51 and NPF-74

Mr. Randall K. Edington
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Arizona Public Service Co.
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Phoenix, AZ 85072-2034

<u>Peach Bottom Atomic Power Station</u>
Exelon Generation Co., LLC
Docket Nos. 50-277 and 50-278
License Nos. DPR-44 and DPR-56

Mr. Michael J. Pacilio
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Warrenville, IL 60555

<u>Perry Nuclear Power Plant</u>
First Energy Nuclear Operating Co.
Docket No. 50-440
License No. NPF-58

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Perry Nuclear Power Plant
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Perry, OH 44081

<u>Pilgrim Nuclear Power Station Unit No. 1</u>
Entergy Nuclear Operations, Inc.
Docket No. 50-293
License No. DPR-35

Mr. Robert Smith
Vice President and Site Vice President
Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5508

<u>Point Beach Nuclear Plant</u>
NextEra Energy Point Beach, LLC
Docket Nos. 50-266 and 50-301
License Nos. DPR-24 and DPR-27

Mr. Larry Meyer
Site Vice President
NextEra Energy Point Beach, LLC
Point Beach Nuclear Plant, Units 1 & 2
6610 Nuclear Road
Two Rivers, WI 54241-9516

Prairie Island Nuclear Generating Plant
Northern States Power Co. Minnesota
Docket Nos. 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Mr. Mark A. Schimmel
Site Vice President
Northern States Power Company - Minnesota
Prairie Island Nuclear Generating Plant
1717 Wakonade Drive East
Welch, MN 55089-9642

Quad Cities Nuclear Power Station
Exelon Generation Co., LLC
Docket Nos. 50-254 and 50-265
License Nos. DPR-29 and DPR-30

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4300 Winfield Road
Warrenville, IL 60555

R. E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
Docket No. 50-244
License No. DPR-18

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Ontario, NY 14519

River Bend Station
Entergy Operations, Inc.
Docket No. 50-458
License No. NPF-47

Mr. Eric W. Olson
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Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61N
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<u>Salem Nuclear Generating Station</u>
PSEG Nuclear, LLC
Docket Nos. 50-272 and 50-311
License Nos. DPR-70 and DPR-75

Mr. Thomas Joyce
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P. O. Box 236
Hancocks Bridge, NJ 08038

<u>San Onofre Nuclear Generating Station</u>
Southern California Edison Co.
Docket Nos. 50-361 and 50-362
License Nos. NPF-10 and NPF-15

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San Onofre Nuclear Generating Station
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San Clemente, CA 92674-0128

<u>Seabrook</u>
NextEra Energy Seabrook, LLC
Docket No. 50-443
License No. NPF-86

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Site Vice President
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NextEra Energy Seabrook, LLC
P.O. Box 300
Seabrook, NH 03874

<u>Sequoyah Nuclear Plant</u>
Tennessee Valley Authority
Docket Nos. 50-327 and 50-328
License Nos. DPR-77 and DPR-79

Mr. Preston D. Swafford
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3R Lookout Place
1101 Market Street
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Shearon Harris Nuclear Power Plant
Carolina Power & Light Co.
Docket No. 50-400
License No. NPF-63

Mr. Christopher L. Burton
Vice President
Progress Energy Carolinas, Inc.
Shearon Harris Nuclear Power Plant
P. O. Box 165, Mail Zone 1
New Hill, NC 27562-0165

South Texas Project
STP Nuclear Operating Co.
Docket Nos. 50-498 and 50-499
License Nos. NPF-76 and NPF-80

Mr. Edward D. Halpin
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STP Nuclear Operating Company
South Texas Project
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Wadsworth, TX 77483

St. Lucie Plant
Florida Power & Light Co.
Docket Nos. 50-335 and 50-389
License Nos. DPR-67 and NPF-16

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700 Universe Boulevard
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Juno Beach, FL 33408-0420

Surry Power Station
Virginia Electric & Power Co.
Docket Nos. 50-280 and 50-281
License Nos. DPR-32 and DPR-37

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<u>Susquehanna Steam Electric Station</u>
PPL Susquehanna, LLC
Docket Nos. 50-387 and 50-388
License Nos. NPF-14 and NPF-22

Mr. Timothy S. Rausch
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769 Salem Boulevard
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Berwick, PA 18603-0467

<u>Turkey Point</u>
Florida Power & Light Co.
Docket Nos. 50-250 and 50-251
License Nos. DPR-31 and DPR-41

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Juno Beach, FL 33408-0420

<u>Vermont Yankee Nuclear Power Station</u>
Entergy Nuclear Operations, Inc.
Docket No. 50-271
License No. DPR-28

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Entergy Nuclear Operations, Inc.
Vermont Yankee Nuclear Power Station
320 Governor Hunt Road
Vernon, VT 05354

<u>Virgil C. Summer Nuclear Station</u>
South Carolina Electric & Gas Co.
Docket No. 50-395
License No. NPF-12

Mr. Thomas D. Gatlin
Vice President Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88, Mail Code 300
Jenkinsville, SC 29065

Vogtle Electric Generating Plant
Southern Nuclear Operating Co.
Docket Nos. 50-424 and 50-425
License Nos. NPF-68 and NPF-81

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Vogtle Electric Generating Plant
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Waynesboro, GA 30830

Vogtle Electric Generating Plant, Units 3 & 4
Southern Nuclear Operating Co.
Docket Nos. 52-025 and 52-026
License Nos. NPF-91 and NPF-92

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Vice President, Regulatory Affairs
Southern Nuclear Operating Company, Inc.
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Bin B022
Birmingham, AL 35242

Waterford Steam Electric Station
Entergy Operations, Inc.
Docket No. 50-382
License No. NPF-38

Ms. Donna Jacobs
Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-0751

Watts Bar Nuclear Plant, Unit 1
Tennessee Valley Authority
Docket No. 50-390
License No. NPF-90

Mr. Preston D. Swafford
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3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

<u>Watts Bar Nuclear Plant, Unit 2</u>
Tennessee Valley Authority
Docket No. 50-391
Construction Permit No. CPPR No. 092

Mr. Michael D. Skaggs
Senior Vice President, Nuclear Generation Development and Construction
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

<u>William B. McGuire Nuclear Station</u>
Duke Energy Carolinas, LLC
Docket Nos. 50-369 and 50-370
License Nos. NPF-9 and NPF-17

Mr. Regis T. Repko
Vice President
Duke Energy Carolinas, LLC
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, NC 28078

<u>Wolf Creek Generating Station</u>
Wolf Creek Nuclear Operating Corp.
Docket No. 50-482
License No. NPF-42

Mr. Matthew W. Sunseri
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P. O. Box 411
Burlington, KS 66839

If you have any questions on this matter, please contact your NRC licensing Project Manager.

Sincerely,

*/ra/*

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Michael R. Johnson, Director
Office of New Reactors

Enclosures:
1. Recommendation 2.1: Seismic
2. Recommendation 2.1: Flooding
3. Recommendation 2.3: Seismic
4. Recommendation 2.3: Flooding
5. Recommendation 9.3: EP
6. Licensees and Holders of Construction Permits

cc: Listserv

Distribution: See next page

**ADAMS Accession Nos.:** ML12056A046 (Pkg.), ML12053A340 (Ltr.) *e-mail concurrence

| | | | | | |
|---|---|---|---|---|---|
| **OFFICE** | PM: NRR/JLD | PM: NRR/JLD | PM: NRR/JLD | LA: NRR/JLD | QTE* |
| **NAME** | GEMiller | JKratchman | CGratton | SRohrer | JDoughtery |
| **DATE** | 03/12/2012 | 03/ 12 /2012 | 03/12/2012 | 03/12/2012 | 02/05/2012 |
| **OFFICE** | BC: NRR/JLD | D: JLD* | OGC* | OD: NRO | OD: NRR |
| **NAME** | RPascarelli | DSkeen | MSpencer | MJohnson | ELeeds |
| **DATE** | 03/11/2012 | 03/11/2012 | 03/11/2012 | 03/ 12 /2012 | 03/ 12 /2012 |

**OFFICIAL RECORD COPY**

**CERTIFICATE OF SERVICE**

I hereby certify that on July 6, 2015, I electronically filed the foregoing with the Clerk of the Court for the United States Court of Appeals for the District of Columbia Circuit by using the appellate CM/ECF system. Participants in the case who are registered CM/ECF users will be served by the appellate CM/ECF system.

By: /s/ Kevin W. Bell
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