POLICY ISSUE
NOTATION VOTE

February 17, 2012 SECY-12-0025

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: PROPOSED ORDERS AND REQUESTS FOR INFORMATION IN RESPONSE TO LESSONS LEARNED FROM JAPAN'S MARCH 11, 2011, GREAT TOHOKU EARTHQUAKE AND TSUNAMI

PURPOSE:

The purpose of this paper is to provide, for Commission consideration, the U.S. Nuclear Regulatory Commission (NRC) staff's proposed orders in response to lessons learned from Japan's March 11, 2011, Great Tōhoku Earthquake and subsequent tsunami. In addition, in accordance with the Staff Requirements Memorandum (SRM) for SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," this paper provides for Commission awareness the requests for information that the staff plans to send to reactor licensees, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 construction permit holders, and combined license (COL) holders as of March 9, 2012. As requested in the October 19, 2011, SRM for SECY-11-0117, "Proposed Charter for the Longer-Term Review of Lessons Learned from the March 11, 2011, Japanese Earthquake and Tsunami," this paper also informs the Commission of the ongoing work conducted under the Charter.

SUMMARY:

The staff proposes to issue three orders. The staff also intends to issue a request for information. These regulatory actions have been informed by stakeholder input from numerous public meetings, recommendations from the Advisory Committee on Reactor Safeguards (ACRS), and the December 2011 Consolidated Appropriations Act, 2012 (Public Law (PL) 112-

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74). The staff has also completed its review of the six additional staff recommendations included in SECY-11-0137 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11272A111), which were beyond those identified in the Near-Term Task Force (NTTF) report (SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated July 12, 2011, ADAMS Accession No. ML11186A950), and that the staff determined had a clear nexus to the Fukushima Dai-ichi event. The staff has developed a process to disposition all subsequent additional issues related to Fukushima Dai-ichi and has applied this process to review the recommendations from the ACRS. The staff has also provided a 6-month status report, which includes the staff's plans to initiate development of a probabilistic risk assessment (PRA) methodology that addresses seismically-induced fires and floods.

BACKGROUND:

In SECY-11-0137, the staff provided its proposed prioritization of the NTTF recommendations in SECY-11-0093 to the Commission. In a December 15, 2011, SRM (ADAMS Accession No. ML113490055), the Commission approved the staff's recommended prioritization, subject to direction provided in SRM-SECY-11-0124, "Staff Requirements-SECY-11-0124 Recommended Actions to be Taken without Delay from the Near-Term Task Force Report," dated October 18, 2011 (ADAMS Accession No. ML112911571).

In SRM-SECY-11-0117, dated October 19, 2011, the Commission also approved the staff's proposed "Charter for the Nuclear Regulatory Commission Steering Committee to Conduct a Longer-term Review of the Events in Japan" (ADAMS Accession No. ML112920034). Among other things, the Charter requires the staff to highlight potential policy issues for the Commission, provide the Commission every 6 months an update on the review work conducted under the Charter, and provide recommendations regarding the sunset of the Steering Committee, the Advisory Committee, and the Japan Lessons Learned Project Directorate.

The staff requirements in SRM-SECY-11-0137, addressed in this paper are the following:

1. Consult with the Commission via notation vote papers before issuing any orders that would lead to a change in the design basis of licensed plants.

2. Inform the Commission 5 business days before issuing letters under 10 CFR Section 50.54(f) associated with the regulatory activities outlined in SECY-11-0137.

3. Inform the Commission of the results of its review of six additional staff recommendations, that went beyond those prepared by the NTTF but which the staff determined had a clear nexus to the Fukushima Dai-ichi event and may warrant additional action. This includes the results of the staff's consideration of filtration of containment vents in the context of the existing Tier 1 issues on hardened reliable vents for boiling-water reactor (BWR) Mark I and Mark II containments.

4. Inform the Commission of how the staff addressed ACRS recommendations, dated November 8, 2011 (ADAMS Accession No. ML11311A264).

5. Initiate a PRA methodology to evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods as part of Tier 1 activities described in SECY-11-0137.

DISCUSSION:

In accordance with the staff's plan for regulatory activities identified as Tier 1 in SECY-11-0137, the staff proposes to issue three orders. The staff also intends to issue a request for information. The staff's approach to implementation of the Tier 1 issues has been enhanced by legislation, ACRS recommendations, stakeholder input, and the review of the additional issues in SECY-11-0137.

Consolidated Appropriations Act, 2012

Section 402 of the December 2011 Consolidated Appropriations Act, 2012 (PL 112-74) provides that:

> 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.

The Conference Report for PL 112-74 states:

> The conferees recognize the progress that the Nuclear Regulatory Commission has made on the recommendations of the Near Term Task Force. Commission staff has proposed a prioritized list of the Task Force recommendations that reflects the order regulatory actions are to be taken. The conferees direct the Commission to implement these recommendations consistent with, or more expeditiously than, the "schedules and milestones" proposed by NRC staff on October 3, 2011. The conferees direct the Commission to maintain an implementation schedule such that the remaining recommendations (not identified as Tier 1 priorities) will be evaluated and acted upon as expeditiously as practicable. The conferees request that the Commission provide a written status report to the House and Senate Committees on Appropriations on its implementation of the Task Force recommendations on the one year anniversary of the Fukushima disaster.

In response to the legislation and input it received from stakeholders, the staff has accelerated the schedule originally proposed in SECY-11-0137, with a goal of issuing the Tier 1 orders and a request for information letter before the first anniversary of Japan's March 11, 2011, Great Tōhoku Earthquake and subsequent tsunami. The staff will provide under separate cover the written status report requested by the conferees.

The staff has also assessed the regulatory activities that will be required to address the "other external hazards" that are referred to in Section 402 of PL 112-74. As stated in the request for information (Enclosure 7), the staff has undertaken a Tier 1 activity to ask licensees to reevaluate seismic, tsunami, and flooding hazards, including the potential for local intense precipitation and site drainage, flooding in streams and rivers, dam breaches and failures, storm surge and seiche, channel migration and diversion and combined effects. Albeit very low, the staff expects that these hazards dominate the risks to the operating fleet of plants from “other external hazards.” As stated in Enclosure 3 and consistent with the prioritization methodology described in SECY-11-0137, the staff proposes to address “other external hazards,” such as wind and missile loads from tornadoes and hurricanes and snow and ice loads for roof design, as a Tier 2 activity that will be initiated as soon as sufficient resources become available.

Stakeholder Participation

To better inform the Tier 1 regulatory actions, the staff conducted over a dozen public meetings with stakeholders to better understand the industry’s current plans and actions, and to obtain stakeholder feedback on the staff’s proposed regulatory actions. Summaries of meetings related to the staff’s near-term actions are available in ADAMS. A list of meeting summaries is provided as Enclosure 1.

The staff also established an e-mail box for members of the public to send input regarding NRC’s resolutions of the Tier 1 recommendations. Comments received as of January 27, 2012, may be found in ADAMS under Accession No. ML12037A220. Comments received on and after January 28, 2012, may be found in ADAMS under Accession No. ML12037A221. The staff has reviewed these comments and considered them in developing the enclosed orders and request for information.

During public meetings in December 2011 and January 2012, and by letter dated December 16, 2011 (ADAMS Accession No. ML11353A008), the industry presented its plans to respond to Fukushima-like events. Industry has developed a concept of a diverse and flexible mitigation capability called “FLEX.” The major principles of FLEX include: (1) adding additional layers of safety to mitigate beyond design bases events, (2) a focus on maintaining key safety functions, (3) multiple supplies of power and cooling water, (4) portable equipment that is reasonably protected, (5) symptom-based guidance and instructions, (6) programmatic controls, and (7) regional support centers. With regard to the details of FLEX, the staff is generally encouraged by the actions being taken by industry in this area. The staff envisions that many elements of FLEX may satisfy the requirements of the order to mitigate challenges to key safety functions resulting from beyond-design-basis natural phenomena hazards (Enclosure 4). The staff will consider additional information about FLEX as it becomes available, in the context of developing implementation guidance for the order requiring development of strategies to deal with beyond-design-basis external events (Enclosure 4). The staff’s regulatory conclusions on the acceptability of FLEX will be based on licensee responses to this order.

Results of Staff Review of Additional Issues Identified in SECY-11-0137

In SECY-11-0137, the staff identified six additional issues that may warrant regulatory action but that were not included with the NTTF recommendations. The staff previously judged these issues to warrant further consideration and potential prioritization based on relative safety significance, nexus to NTTF recommendations, and other ongoing staff activities. As directed by SRM-SECY-11-0137, the staff conducted an assessment of whether the issues should be included with the Japan lessons-learned activities and determined if any regulatory action is recommended or necessary. The staff applied the same prioritization process described in SECY-11-0137. The result of the staff’s assessment is provided in Enclosure 2.

The staff has determined that some of the additional issues should be included in existing Tier 1 activities. In accordance with the direction in SRM-SECY-11-0137, the additional issue of filtration of containment vents was merged with the Tier 1 issue of hardened vents for Mark I and Mark II containments such that further analysis and interaction with stakeholders will inform whether filtered vents should be required. The staff has determined that consideration of severe accident conditions in the design and operation of the vent, the addition of filters to hardened reliable vents, and consideration of vents in areas other than primary containment, will be the topic of a policy paper to the Commission in July 2012.

The staff believes that the requirements for hardened reliable vents in the proposed order (Enclosure 5) are important to ensure core and containment cooling, and that these requirements should be imposed before the staff completes its evaluation of the technical and policy issues associated with imposing additional requirements, as described above. In public meetings, the staff has encouraged licensees to consider the potential for the later addition of filters. However, the industry has stated that the addition of filters to hardened containment vents may require modifications to the vent design. In light of this, a consideration in the staff's proposal to issue the proposed order now is that the proposed order requires submission of integrated plans for implementing the requirements of the order by February 28, 2013, eight months after the staff plans to send the July 2012 policy paper to the Commission for consideration. As a result, licensees should have time to revise draft plans in response to any new Commission direction before the integrated implementation plans are due.

The staff also assessed the issue of loss of the ultimate heat sink function to be of sufficient safety significance as to warrant inclusion with the ongoing Tier 1 regulatory actions to mitigate or prevent challenges to key safety functions resulting from seismic and flooding hazards. Additionally, a potential loss of ultimate heat sink function due to other natural external hazards will be considered as part of a new Recommendation 2.1 Tier 2 item, which will address reevaluation of other natural external hazards for each facility.

The additional issue of instrumentation for seismic monitoring has been transferred from the Japan lessons-learned process and will be further considered under the ongoing action plan for the August 2011 Central Virginia earthquake.

The remaining three additional issues (emergency planning zone size, prestaging potassium iodide beyond 10 miles, and transferring spent fuel to dry cask storage) have been prioritized as Tier 3 items. The staff has determined that the current regulatory approaches to these issues are acceptable. The staff will review new information that becomes available as a result of specific ongoing activities to confirm this conclusion and gain additional insights. The staff will further address these Tier 3 recommendations in its paper scheduled to be sent to the Commission in July 2012.

Results of Staff Review of ACRS Recommendations and Other Additional Issues

The staff developed a process to disposition all additional issues. A description of the staff's process and the results of its evaluation of the ACRS recommendations are provided in Enclosure 3. The staff's evaluation of other additional stakeholder recommendations is an ongoing process. The staff plans to make available the results of its evaluation of these issues on the NRC's public Web site. By letter dated February 15, 2012 (ML12046A145), the ACRS provided additional recommendations, which the staff will address through its additional issues process.

Proposed Orders

Consistent with its recommendations in SECY-11-0137, the staff proposes to issue three orders. Two orders are proposed to be issued to all reactor licensees, including holders of active or deferred construction permits[1] under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and holders of combined licenses (COLs)[2] under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," regarding: (1) development of strategies to mitigate beyond design basis natural phenomena which addresses both multi-unit events and reasonable protection of equipment identified under such strategies, and (2) installation of enhanced spent fuel pool instrumentation. The third order pertaining to reliable containment vents is proposed to be issued to licensees operating BWRs with Mark I and Mark II containments. Each of the orders is focused on enhancing defense in depth at nuclear power plants through increased capabilities to minimize the potential for core damage following a beyond design basis external event. In order to effectuate timely implementation, each order has been made immediately effective. In addition, pursuant to 10 CFR 2.202, the NRC finds that the public health, safety and interest require that these Orders be made immediately effective.

The licensing approach for operating power reactors in all three orders is similar. The staff plans to prepare guidance for implementing the technical requirements of the orders by August 2012. Licensees will then be required, by February 28, 2013, to submit to the Commission for review an overall integrated plan including a description of how compliance with the requirements of the order will be achieved. After reviewing the licensee's submittals, the staff plans to issue facility-specific orders imposing license conditions that address the requirements of the orders. Licensees are required to provide an initial status report within 60 days of the issuance of the staff's guidance, and additional reports every 6 months following the submittal of the overall integrated plans. The purpose of the status reports is to ensure that staff can monitor licensees' incremental progress and take appropriate regulatory action, if needed. Each licensee will be required to achieve full compliance within two refueling outages after submittal of its overall integrated plan, or by December 31, 2016, whichever comes first.

Adequate Protection

As stated in the enclosed orders, to protect public health and safety from the inadvertent release of radioactive materials, the NRC's defense-in-depth strategy includes multiple layers of protection: (1) prevention of accidents by virtue of the design, construction and operation of the plant, (2) multiple mitigation features to prevent radioactive releases should an accident occur, and (3) emergency preparedness programs that include measures such as sheltering and evacuation. The defense-in-depth strategy also provides for multiple physical barriers to contain the radioactive materials in the event of an accident. The barriers are the fuel cladding, the reactor coolant pressure boundary, and the containment. These defense-in-depth features are embodied in the existing regulatory requirements and thereby provide adequate protection of

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[1] Bellefonte Nuclear Plant, Units 1 and 2 (Construction Permit Numbers CPPR-122 and CPPR-123); and Watts Bar Unit 2 (CPPR-92).

[2] Vogtle Electric Generating Plant Units 3 and 4 (NPF-91 and NPF-92)

the public health and safety. However, the events at Fukushima highlighted the possibility that extreme natural phenomena could challenge the prevention, mitigation, and emergency preparedness defense-in-depth layers.

Accordingly, in the enclosed orders, the staff is proposing to redefine the level of protection regarded as adequate pursuant to 10 CFR 50.109(a)(4)(iii) and require actions of licensees to meet that new level of protection. A summary of the staff's justification for redefining the level of protection regarded as adequate for each of the orders is provided below.

An order requiring development of strategies to deal with beyond-design-basis external events resulting in simultaneous loss of all ac power and loss of normal access to the ultimate heat sink is provided as Enclosure 4. The events at Fukushima highlighted the possibility that extreme natural phenomena could challenge the prevention, mitigation, and emergency preparedness defense-in-depth layers. The strategies and guidance developed and implemented by licensees in response to the requirements imposed by this order will provide the necessary capabilities to supplement those of the permanently installed plant structures, systems, and components that could be unavailable following beyond-design-basis external events. These strategies and guidance will enhance the safety and preparedness capabilities established following the events of September 11, 2001, and codified as 10 CFR 50.54(hh)(2). In order to address the potential for more widespread effects of beyond-design-basis external events, this order requires licensees to have increased capabilities to implement multiple strategies concurrently at multiple units on a site. The strategies shall be developed to add multiple ways to maintain or restore core cooling, containment and SFP cooling capabilities in order to improve the defense in depth of licensed nuclear power reactors.

With regard to the order requiring reliable, hardened vents in BWR Mark I and Mark II containments (Enclosure 5), the events at Fukushima Dai-ichi highlight the possibility that beyond-design-basis external events could challenge the prevention, mitigation and emergency preparedness defense-in-depth layers. At Fukushima, limitations in time and unpredictable conditions associated with the accident significantly challenged the attempts by the responders to preclude core damage and containment failure. In particular, the operators were unable to successfully operate the containment venting system. The inability to reduce containment pressure inhibited efforts to cool the reactor core. Had additional backup or alternate sources of power been available to operate the containment venting system remotely, or had certain valves been more accessible for manual operation, the operators at Fukushima might have been able to depressurize the containment earlier. This, in turn, could have allowed operators to implement strategies using low pressure water sources. Thus, the events at Fukushima demonstrate that reliable hardened vents at BWR facilities with Mark I and Mark II containment designs are important to maintain core and containment cooling.

Finally, Enclosure 6 to this paper contains an order requiring enhanced spent fuel pool (SFP) instrumentation. During the events in Fukushima, responders were without reliable instrumentation to determine the water level in the SFP. This caused concerns that the pool may have boiled dry, resulting in fuel damage. Fukushima demonstrated the confusion and misapplication of resources that can result from beyond-design-basis external events when adequate SFP instrumentation is not available. The SFP level instrumentation at U.S. nuclear power plants is typically narrow range and, therefore, only capable of monitoring normal and slightly off-normal conditions. Although the likelihood of a catastrophic event affecting nuclear power plants and the associated SFPs in the United States remains very low, beyond-design-basis external events could challenge the ability of existing spent fuel pool instrumentation in providing emergency responders with reliable information on the condition of SFPs. Reliable and available indication is essential to ensure plant personnel can effectively prioritize emergency actions.

The staff continues to affirm that current regulatory requirements and existing plant capabilities allow the NRC to conclude that a sequence of events like the Fukushima Dai-ichi accident is unlikely to occur in the United States. Therefore, continued operation and continued licensing activities do not pose an imminent threat to public health and safety. However, the NRC's assessment of new insights from the events at Fukushima Dai-ichi leads the staff to conclude that additional requirements should be imposed on licensees to increase the capability of nuclear power plants to mitigate beyond-design-basis external natural events. The staff considers that all nuclear power plants should be at the redefined level of adequate protection by December 31, 2016, at the latest.

Should the Commission find that the staff's evaluation does not support a finding or declaration that the proposed orders involve redefining the level of protection to the public health and safety or common defense and security that should be regarded as adequate, the Commission may administratively exempt these orders from applicable backfit requirements. The Commission took this extremely rare action in its issuance of the Aircraft Impacts final rule (74 FR 28112 (July 9, 2009)). If the Commission chooses this course, the orders would need to be revised to provide a well articulated explanation for invoking this exemption.

Requests for Information

As required in SRM-SECY-11-0137, the staff is informing the Commission at least 5 business days before issuing letters associated with the regulatory activities outlined in SECY-11-0137 (Enclosure 7). The enclosed letter addresses seismic and flooding reevaluations (Recommendation 2.1), seismic and flooding hazard walkdowns (Recommendation 2.3) and a request for licensees to assess their current communications system and equipment under conditions of onsite and offsite damage and prolonged station blackout (SBO) and perform a staffing study to determine the number and qualifications of staff required to fill all necessary positions in response to a multi-unit event (Recommendation 9.3). As stated above, the staff has prioritized as a new Tier 2 activity to continue stakeholder interactions on development of additional requests for information that will address licensee reevaluations of external hazards other than seismic, tsunami and flooding against current applicable Commission requirements and guidance (Enclosure 3).

The staff will request information from COL holders, active and deferred construction permit holders and holders of operating reactor licenses in accordance with provisions of Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended (the Act). These provisions of the Act are implemented for holders of operating reactor licenses issued under 10 CFR Part 50 in 10 CFR 50.54(f). For 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 will not be requested to respond to those portions of the 10 CFR 50.54(f) letter. Similarly, information requests related to walkdowns are not applicable to holders of construction permits under 10 CFR Part 50. Operating power reactor licensees under 10 CFR Part 50 will be requested to respond to all of the information requests provided in Enclosure 7 to this paper.

Under 10 CFR 50.54(f), when information is not sought to verify compliance with a facility's current licensing basis, the staff is required to prepare a reason or reasons for each information request prior to issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information. As noted in the body of the enclosed letter, protection of plants from natural phenomena is critical for continued safe operation of nuclear power plants. Given that new information has been developed on natural phenomena hazards since the licensing basis of the operating plants was established, the staff finds that it is necessary to confirm the adequacy of the hazards assumed for U.S. plants and their ability to protect against them. Further, the staff finds that the accident at Fukushima highlights a need to verify the adequacy of emergency planning to address a prolonged SBO and multiunit events. Finally, the reevaluation and related information analysis will serve to meet the NRC's obligation under the Consolidated Appropriations Act, for 2012 (PL 112-74), Section 402.

The Office of Information Services is currently seeking expedited approval from the Office of Management and Budget (OMB) for the industry burden to respond to the requests for information. The staff will continue to work with OMB to meet the requirements of the paperwork reduction act for information collection.

Definition of Vulnerability

In SRM-SECY-11-0124, the Commission directed the staff to define "vulnerability," in the context of the staff's requests for information regarding actions that licensees have taken, or have planned to take, to address plant-specific vulnerabilities associated with the reevaluation of seismic and flooding hazards. In the staff's request for information (Enclosure 7), the staff defined plant-specific vulnerabilities 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.

The definition is broad enough to capture both prevention and mitigation aspects and also includes features of protection such as hardware, procedures, temporary measures, and potentially available off-site resources. This definition allows 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.

Immediate NRC and Industry Actions

The initial response of the NRC and industry to the nuclear reactor accident at Fukushima Dai-ichi was to perform an immediate assessment of domestic nuclear power plants. The NRC issued an information notice, a bulletin, and two temporary instructions which directed NRC inspectors to accomplish the following:

- Confirm the reliability of licensees' strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire.

- Inspect the readiness of nuclear power plant operators to implement severe accident management guidelines.

The NRC inspections were completed by April 15, 2011. The minor or low safety significance issues that were identified posed no imminent threat to public health and safety. Identified issues have been entered into licensee corrective action programs.

In addition and in parallel with NRC actions, the Institute of Nuclear Power Operations (INPO) informed NRC staff that it had asked nuclear power plant licensees to accomplish the following:

- Verify the capability to mitigate internal and external flooding events required by station design.
- Perform walkdowns and inspections of important equipment needed to mitigate fire and flood events to identify the potential that the equipment's function could be lost during seismic events and develop mitigating strategies for identified vulnerabilities.
- Increase sensitivity to spent fuel storage event response and ensure that a high state of readiness is maintained to respond to events that challenge spent fuel storage integrity.
- Develop plant specific information concerning coping times and design limitations for extended loss of power events.

Status update on other Charter activities

The charter requires the staff to provide the Commission every six months an update on the review work conducted under the Charter, highlight potential policy issues for the Commission, and provide recommendations regarding the sunset of the Steering Committee, Advisory Committee, and the Japan Lessons Learned Project Directorate. The staff's first 6-month summary is provided as Enclosure 8. This includes, as required in SRM-SECY-11-0137, a resource estimate and schedule for development of a probabilistic risk assessment (PRA) methodology to implement NTTF Recommendation 3, which is to identify potential enhancements to the capability to prevent or mitigate seismically-induced fires and floods.

New Reactors and other NRC-regulated facilities

Design Certifications and Combined Licenses

For design certifications and combined license applications submitted under 10 CFR Part 52 that are currently under active staff review, the staff plans to assure that the Commission-approved Fukushima actions are addressed prior to certification or licensing. To date, the staff has met with AREVA and MHI to understand their plans for incorporating changes into their respective designs to effectively address the design-related Fukushima items. The staff will also request all COL applicants to provide the information required by the orders and request for information letters described in this paper, as applicable, through the review process. New reactor and operating reactor staff are coordinating their regulatory positions to assure that the resolutions proposed by new reactor design certification and combined license applicants are not in conflict with those proposed and accepted by the staff for operating reactors.

For new reactor design certification or license applications (e.g., construction permit, operating license, combined license) not yet submitted, the staff expects those applicants to address the

Commission-approved Fukushima actions in their applications, prior to submittal, to the fullest extent practicable.

On February 10, 2012, the NRC issued COLs for the Vogtle Electric Generating Plant Units 3 & 4. Also pending before the Commission are COLs for the Virgil C. Summer Nuclear Station Units 2 & 3. These COLs reference the AP1000 Design, which was recently certified by the Commission in Appendix D to Part 52. Consistent with the “Policy Statement on Regulation of Advanced Reactors,” (73 FR 60612, October 14, 2008), the AP1000 design has enhanced safety features and safety margins beyond those contained in the licensing bases for current operating reactors. These design features and safety margins translated into enhanced operational strategies for the COLs. The applicable Commission-approved Fukushima actions not already addressed as part of the licensing process will be addressed in the same manner as operating reactor licensees. Specifically, the 50.54(f) letter being sent to operating reactors (Enclosure 7) will also be sent to Vogtle to address Tier 1 Recommendation 9.3 in its entirety.

The staff is not requesting Vogtle to respond to Tier 1 Recommendation 2.1 or 2.3. Tier 1 Recommendation 2.1 requests that licensees reevaluate the seismic and flooding hazards for their sites against present-day NRC requirements and guidance. As discussed in the 50.54(f) letter, a new seismic source characterization model (NUREG-2115, “Central and Eastern United States Seismic Source Characterization for Nuclear Facilities”) has recently been issued. This new model was not available to Vogtle during the development of its COL application, and the applicant used an NRC-endorsed source model that had been recently updated. As discussed as part of NRC staff testimony at the COL hearings for Vogtle, the staff believes that use of the new source model would not result in differences in the seismic hazard characterizations that would affect the plant design for this site. The NRC staff continues to maintain this position and therefore considers that Recommendation 2.1 has been addressed as part of the completed COL reviews. Once the computer software becomes available, the staff will confirm this position by developing seismic hazard curves for each of the sites using the new source model. Tier 1 Recommendation 2.3 is not applicable to a facility that has not yet been constructed.

The staff also proposes to order Vogtle to address the portions of Tier 1 Recommendations 4.2 and 7.1 not already covered by the referenced certified design or COL review. With regard to Recommendation 4.2 for mitigation strategies for beyond-design-basis external events, the AP1000 standard design includes passive design features that provide core, containment and spent fuel pool cooling capability for 72 hours, without reliance on AC power. These features do not rely on access to any external water sources since the containment vessel and the passive containment cooling system serve as the safety-related ultimate heat sink. The AP1000 design also includes equipment to maintain required safety functions in the long term (beyond 72 hours to 7 days). Connections are provided for generators and pumping equipment that can be brought to the site to back up the installed equipment. The staff concluded in its final safety evaluation report for the AP1000 design that the installed equipment (and alternatively, the use of transportable equipment) is capable of supporting extended operation of the passive safety systems to maintain required safety functions in the long term. The proposed order requires Vogtle, prior to fuel load, to address requirements for mitigation strategies to sustain core cooling, containment and SFP cooling capabilities functions indefinitely.

With regard to Recommendation 7.1 for SFP level indication, the AP1000 standard design includes two permanently fixed safety related level instruments with the capability for a third instrument connection. The instrumentation range covers the top of the pool to the top of the fuel racks. The safety related classification ensures seismic qualification consistent with the SFP design, independence of instrument channels and power supplies, and routine testing and calibration. The proposed order requires Vogtle to provide additional design information to ensure missile and falling debris protection, equipment qualification for extended water saturation conditions, display indications, and the capability to connect portable power supplies to the instrumentation.

Fuel Cycle Facilities

On September 30, 2011, the staff issued and initiated temporary instruction (TI) 2600/015, “Evaluation of Licensee Strategies for the Prevention and/or Mitigation of Emergencies at Fuel Facilities” (ADAMS Accession No. ML111030453). These inspection activities include all 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material,” licensees with an integrated safety analysis (ISA), all 10 CFR Part 40, “Domestic Licensing of Source Material,” licensees with a license-required ISA, and all 10 CFR Part 76, “Certification of Gaseous Diffusion Plants,” certificate-holders currently in operation. The inspection activities are ongoing and are to be completed within one year of issuance of the TI. The staff will evaluate any findings under this TI using its normal inspection processes. As stated in SECY-11-0137, the staff will continue to evaluate the applicability of lessons learned to licensed facilities other than power reactors (e.g., research and test reactors, independent spent fuel storage installations, and reactors that have permanently ceased operations but still maintain fuel in a spent fuel pool), and take appropriate actions.

COMMITMENTS:

As stated in SECY-11-0137, the staff will provide in July 2012 an evaluation of the Tier 3 recommendations. In addition, in SRM-SECY-11-0137, the Commission directed the NRC staff to take certain actions and provided further guidance including directing the staff to consider filtered vents. The staff has determined that there are technical and policy issues that need to be considered before any regulatory action can be taken to require licensees to install filtered vents. This issue will require further examination of other important policy matters related to the treatment of severe accidents, including filtration. The staff will present these policy matters in its July 2012 paper. The staff will also promptly inform the Commission of any additional recommendations that are prioritized as Tier 1.

RECOMMENDATION:

The staff recommends the Commission approve issuance of the proposed orders. In order to support issuance of the orders by the March 11, 2012, anniversary of the events in Japan, the staff requests Commission approval by March 2, 2012.

RESOURCES:

In fiscal year (FY) 2012 and FY 2013, the staff will reallocate from existing resources to start new Tier 1 and 2 activities described in this paper. This reallocation is less than the 4 full-time equivalent (FTE) and $500,000 that requires Commission approval.

Previously, SECY-11-0137 described the Tier 1 and 2 activities and had an estimate of 30 FTE in FY 2012 and 90 FTE in FY 2013.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection. The Office of the Chief Financial Officer has reviewed this paper for resource implications and has concurred. The request for information (Enclosure 7) has been reviewed by the Committee on Review of Generic Requirements, which endorsed this regulatory product with minor editorial comments.

/RA/

R. W. Borchardt
Executive Director
for Operations

Enclosures:
1. Public Meetings related to Japan Lessons-Learned
2. Disposition of Additional Recommendations from SECY-11-0137
3. Disposition of ACRS Recommendations
4. Order on Mitigating Strategies for Beyond-Design-Basis External Events
5. Order on Reliable, Hardened, and Filtered Vents (Mark I and II BWRs)
6. Order on Spent Fuel Pool Instrumentation
7. Draft 50.54(f) letter - External Hazards Reevaluation, Walkdown and Emergency staffing
8. 6-month Status Update on other Charter Activities

Public Meetings Related to Tier 1 Japan Lessons-Learned Regulatory Actions

Meeting Dates (first) (second)	Meeting Purpose	Summaries[1]
October 7, 2011 February 9, 2012	**Meetings of the Advisory Committee on Reactor Safeguards**	ML11290A192 ML12046A145
November 29, 2011	**Commission Meeting with Advisory Committee on Reactor Safeguards**	ML11345A000
December 1, 2011	**U.S. Nuclear Regulatory Commission (NRC) and Industry Joint Steering Committee** on strategies to implement the Near-Term Task Force (NTTF) recommendations.	ML11341A160
December 8, 2011 January 18, 2012	**NTTF Recommendation 4.2** on issuing orders to licensees to mitigate challenges to key safety functions following extreme natural phenomena events regulations currently provided in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(hh)(2) to provide reasonable protection for equipment from the effects of design-basis external events and to add equipment as needed to address multiunit events.	ML11348A098 ML12032A044
December 12, 2011 January 9, 2012 January 19, 2012	**NTTF Recommendation 9.3** on staffing and communications in response to emergencies. The meetings focused on a general approach and introduction to the implementation of recommendations under consideration.	ML11349A008 ML12032A221 ML12033A118
December 14, 2011 January 18, 2012	**NTTF Recommendations 2.1 and 2.3**, on flooding and seismic protections. The meetings focused on a general approach and introduction to the implementation of this recommendation.	ML11353A390 ML11356A230
December 15, 2011 January 19, 2012	**NTTF Recommendation 7.1**, on spent fuel pool instrumentation. The meetings focused on a general approach and introduction to the implementation of this recommendation.	ML11356A061 ML11361A043
December 15, 2011 January 17, 2012	**NTTF Recommendation 5.1**, on reliable and hardened vents. The meetings focused on a general approach and introduction to the implementation of this recommendation.	ML12038A245 ML12025A020
January 13, 2012	**NRC and Industry Joint Steering Committee**	ML11362A202

[1] Accession numbers in the Agencywide Documents Access and Management System

STAFF ASSESSMENT AND PRIORITIZATION OF ADDITIONAL ISSUES IDENTIFIED IN SECY-11-0137

As directed by Staff Requirements Memorandum (SRM)-SECY-11-0137, “Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned,” dated October 3, 2011, the staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed the additional issues identified in SECY-11-0137 within the context of the NRC’s existing framework and considered whether to recommend any additional regulatory action. A team consisting of NRC senior management representatives and technical experts conducted this review. The staff used the same prioritization process that was used in SECY-11-0137. The staff’s prioritization and assessment process generally prioritized the additional issues into either Tier 1 or Tier 3 as defined in SECY-11-0137.

The first tier consists of those additional issues that the staff determined should be started without unnecessary delay and for which there is sufficient resource flexibility, including the availability of critical skill sets. The Tier 1 issues are the following:

- filtration of containment vents
- loss of ultimate heat sink

However, a portion of the loss of ultimate heat sink issue, related to the impact of external natural hazards other than flooding hazards, will be addressed by a new Recommendation 2.1 Tier 2 action. This new Tier 2 action, which will be initiated when sufficient critical skill sets become available, is discussed further in Enclosure 3.

The third tier consists of those additional issues that require further staff study. Depending on the outcome of long-term studies, the staff may recommend additional regulatory actions. The staff has focused its initial efforts on developing the assessment, schedules, milestones, and resources associated with the additional Tier 1 and Tier 2 activities. Hence, information regarding the Tier 3 additional issues is not included in this enclosure. The staff is currently developing an evaluation of the Tier 3 additional issues, which will be included in a paper due to the Commission in July 2012. The Tier 3 additional issues are as follows:

- basis of emergency planning zone size (long-term study)
- pre-staging of potassium iodide beyond 10 miles (long-term study)
- transfer of spent fuel to dry cask storage (long-term study)

The additional issue of “Instrumentation for Seismic Monitoring” has been transferred from the Japan Lessons-Learned process and will be further considered under the ongoing action plan for the August 2011 Central Virginia Earthquake and thus is not being prioritized in this paper.

This enclosure assesses the Tier 1 additional issues in the order listed above. The NRC concluded that the Tier 1 additional issues are of sufficient safety significance that the staff should proceed to consider them without delay, and it has already included them in its ongoing work on Tier 1 activities.

Tier 1 – Filtration of Containment Vents

The staff is considering requiring the filtration of containment vents to reduce the spread of radioactive contamination during a beyond-design-basis event.

Regulations and Guidance

1. General Design Criterion (GDC) 16, "Containment Design," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that the reactor containment and associated systems "be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

2. Under 10 CFR 50.34(f)(3)(iv), "Contents of applications; technical information – Additional TMI-related requirements," requires each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982, must provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. This requirement only applied to the small number of applications that were pending as of February 16, 1982.

Staff Assessment and Basis for Prioritization

The Fukushima Dai-ichi event highlighted the importance of the primary containment heat removal function in boiling-water reactor (BWR) accident response. In particular, it showed the importance of accessibility of the valves, which are required to open and close the vent independent of alternating current power. As directed by the Commission in SRM-SECY-11-0137, the staff has prioritized the issue of filtration of containment vents as a Tier 1 issue.

The staff has determined that there are technical and policy issues to be resolved before regulatory action can be taken to require licensees to install filtered vents. One policy issue that needs further study is whether containment vents, with or without filters, should be required to operate under severe accident conditions. The staff will also take into consideration regulatory action to require controlled venting of structures other than the reactor building, such as those housing spent fuel pools. The staff plans to provide the Commission a notation vote paper on these policy issues in July 2012.

At this time, the staff is proposing regulatory action to require that all operating BWR facilities with Mark I and Mark II containments have a reliable hardened venting capability, without filters, for events that can lead to core damage. In public meetings, the staff has encouraged licensees to consider the potential for the later addition of filters.

Staff Activities

The staff, as a near-term action, is currently undertaking regulatory activities to consider filtered vents for BWR reactor facilities with Mark I and Mark II containments, and present to the Commission a notation vote paper outlining any policy issues and a recommendation for regulatory action.

Unique Implementation Challenges

The staff recognizes that several technical and policy issues need to be considered before a decision is made on whether filters should be required, such as whether containment vents with or without filters need to be operable in severe accidents, and whether structures other than the reactor building should be required to have controlled venting.

Schedules and Milestones

The schedule and milestones previously described in SECY-11-0137 for ongoing Tier 1 activities are not expected to change with the addition of this item. The staff will provide the Commission with a notation vote paper describing policy issues in July 2012.

Resources

The resources previously described in SECY-11-0137 for ongoing Tier 1 activities are not expected to change at this time. The staff will include in the scheduled July 2012 Commission paper an estimate of any additional resources beyond those provided in SECY-11-0137 needed for regulatory action to address the policy issues described above.

Tier 1 – Loss Ultimate Heat Sink

The staff has evaluated the implications of a loss of ultimate heat sink (UHS) at U.S. nuclear power plants and determined the regulatory actions needed in this area.

Regulations and Guidance

1. GDC 44, "Cooling Water," states that a system shall be provided to transfer heat from structures, systems, and components important to safety, to an UHS. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

2. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 2, issued January 1976, describes a basis acceptable to the NRC staff that may be used to implement GDC 44 and 2, "Design Bases for Protection Against Natural Phenomena," with regard to a particular feature of the cooling water system: specifically, the UHS, including single-failure criteria and the overall capacity of the UHS.

3. Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated July 18, 1989, details the possible need for surveillance and control programs to reduce the incidence of service water system fouling, based on an analysis of operating experience.

4. NUREG-1275, Volume 3, "Operating Experience Feedback Report – Service Water System Failures and Degradations," was issued November 1988.

5. GL 91-13, "Request for Information Related to the Resolution of Generic Issue 130, 'Essential Service Water System Failures at Multi-Unit Sites,'" dated September 19, 1991, requested information from seven sites seen as at high risk for a loss-of-service-water initiating event based on their configuration following evaluation of Generic Issue 130.

6. NUREG/CR-5526, "Analysis of Risk Reduction Measure Applied to Shared Essential Service Water Systems at Multi-Unit Sites," was issued June 1991.

7. Under 10 CFR 50.54(hh)(2), licensees must develop and implement guidance and strategies to maintain or restore core cooling capabilities under the circumstances associated with the loss of a large area of the plant caused by explosions or fire.

Staff Assessment and Basis for Prioritization

As a result of the March 11, 2011, Great Tōhoku Earthquake and subsequent tsunami, Fukushima Dai-ichi, Units 1-3, lost the capacity to release decay heat to the ultimate heat sink (the ocean). In many plants, both foreign and domestic, an adjacent body of water is used as a heat sink for main circulating water, providing cooling for steam exiting the main condenser, and also as a heat sink for the service water system. The event at Fukushima Dai-ichi reinforces the need not only to evaluate the capacity to restore an ultimate heat sink promptly under accident conditions, but also to consider, in accident planning, an alternative means for maintaining reactor stability in a hot-standby condition for an extended period of time when normal modes of heat transport to the UHS are unavailable.

Though loss of service water is analyzed in risk models as an initiating event, a complete loss of service water is not considered in accident analysis, and plants are not typically designed to be able to cope with an extended loss of service water or the UHS. Depending on plant-specific emergency response capabilities, failure to recover service water cooling via the UHS has a high probability of leading to core damage, on the order of 10 percent for some plants.

Potential causes for loss of service water have been addressed by the NRC multiple times over the past 30 years, most notably in GL 89-13. Though measures taken by industry to reduce risk exposure, both in response to the GL, and on a voluntary basis, have addressed many potential causes of a loss of UHS, new failure modes continue to present themselves.

Ongoing regulatory activities following the Fukushima Dai-ichi accident are addressing several aspects of the loss of UHS. Tier 1 activities for seismic and flood reevaluations and walkdowns will address protection of the UHS systems (Near-Term Task Force (NTTF) Recommendations 2.1 and 2.3). A new Tier 2 Recommendation 2.1 item on other natural external hazards will also address protection of the UHS systems. Tier 1 activities for station blackout mitigating strategies (NTTF Recommendation 4.1) and mitigation of beyond-design-basis natural phenomena events (NTTF Recommendation 4.2) will also include regulatory actions for licensees to provide strategies for mitigating a loss of access to the normal UHS. The staff has established that the term “UHS systems” is intended to include loss of the cooling media, loss of the ability to pump the cooling media, loss of heat exchangers and combinations of losses, while the access to UHS is all of the above with the exception of the cooling media.

The staff concludes that this issue would improve safety. Since sufficient resource flexibility, including availability of critical skill sets, exists, the staff prioritized this action as a Tier 1 issue.

Staff Activities

The staff, as a near-term action, is currently undertaking regulatory activities to do the following:

1. Request that licensees include UHS systems in the reevaluation and walkdowns of site-specific seismic and flooding hazards using the methodology described in SECY-11-0137, and identify actions that have been taken, or are planned, to address plant-specific issues associated with the updated seismic and flooding hazards in conjunction with the resolution of NTTF Recommendations 2.1 and 2.3.

2. Incorporate the loss of UHS as a design assumption in the resolution of station blackout rulemaking activities in conjunction with the resolution of NTTF Recommendation 4.1.

3. Order licensees to provide mitigating measures for beyond-design-basis external events to also include a loss of access to the normal UHS in conjunction with the resolution of NTTF Recommendation 4.2.

4. Request licensees to include UHS systems in the reevaluation of site-specific natural external hazards, and identify actions that have been taken, or are planned, to address plant-specific issues associated with the updated hazards in conjunction with the resolution of the new Tier 2 Recommendation 2.1 activity described in Enclosure 3, “Other Natural External Hazards.”

Unique Implementation Challenges

In order to address the new Tier 2, Recommendation 2.1, activity described in Enclosure 3, “Other Natural External Hazards,” the staff recognizes that the NRC and industry have limited, specialized expertise (e.g., physical scientists, hydrologists) to complete the actions associated with this recommendation.

Schedules and Milestones

The schedule and milestones previously described in SECY-11-0137 for ongoing Tier 1 activities are not expected to change with the addition of this item.

Resources

The resources previously described in SECY-11-0137 for ongoing Tier 1 activities are not expected to change with the addition of this item.

STAFF'S PRIORITIZATION OF ACRS RECOMMENDATIONS FOR NRC ACTIONS TO BE TAKEN IN RESPONSE TO FUKUSHIMA LESSONS–LEARNED

The purpose of this enclosure is to provide the results of the U.S. Nuclear Regulatory Comission (NRC) staff's analysis of recommendations made by the Advisory Committee on Reactor Safeguards (ACRS) in letters dated October 13, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11284A136), and November 8, 2011 (ADAMS Accession No. ML11311A264). This enclosure also describes the staff's process for resolving the ACRS recommendations, as well as any other Fukushima–related issue that arises from the staff's ongoing lessons-learned deliberations, stakeholder interactions, and international outreach activities.

Process for Addressing Additional Issues

The staff developed a process to disposition all additional issues, including recommendations by the ACRS. All issues are reviewed by a panel of senior-level advisors from different NRC program offices. The panel determines whether each issue represents a valid safety concern, and whether there is a clear nexus to the Fukushima Dai–ichi accident. If neither criterion is met, or only one criterion is met, the panel chooses to either disposition the issue with no action, or direct it to one of the NRC's existing regulatory processes (e.g., generic issue process). If both criteria are met, the issue is forwarded for further consideration by the cognizant technical staff in the appropriate NRC line organization. Should the issue go forward, the cognizant technical staff is tasked with developing a proposal for Steering Committee (SC) disposition. The SC may elect to take no further action, disposition the issue using an existing NRC process, or prioritize the issue as a Tier 1, 2, or 3 item under the Japan Lessons–Learned Program.

This process will be used to disposition recommendations and issues sent to the NRC. The SC is routinely presented with a list of issues screened out by the panel of senior–level advisors for review, and it ultimately determines the final prioritization and disposition of each issue. Once this occurs, the staff documents the SC's findings, in detail, and plans to publish the results on the NRC's public Web site.

ACRS Recommendations

The staff has evaluated the recommendations of the ACRS in its October 13, 2011, and November 8, 2011, letters, using the staff's process for screening additional recommendations. The staff documented the SC's disposition of each ACRS recommendation, and has ensured that the cognizant technical staff working groups have used them to enhance the Tier 1, 2, and 3 actions that will be taken as a result of the events at the Fukushima Dai–ichi Nuclear Power Plant. A summary of the staff's disposition of the ACRS recommendations is provided in the table below. The staff addressed ACRS Recommendations 1(a)-1(g), 2(a)-2(f), and 3 from the letter dated October 13, 2011; as well as ACRS Conclusions 1-5 from letter dated November 8, 2011.

The staff also acknowledges the receipt of ACRS letter dated February 15, 2012. The staff will evaluate these additional ACRS comments/ recommendations and will enter them into its process for screening additional recommendations decribed above.

ACRS Recommendations Incorporated into Tier 1 Activities	
ACRS Recommendation	**Staff Response**
• ACRS Recommendation 1(b)—"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." • ACRS Conclusion 2—"Tier 1 recommendations should be expanded to include the additional immediate actions recommended in our October 13, 2011, report, regarding flooding hazard reevaluations, integrated walkdowns, station blackout, boiling water reactor (BWR) hardened vents, shared ventilation systems, hydrogen control and mitigation, spent fuel pools (SFPs) and integration of onsite emergency actions."	The NRC staff expanded NTTF Recommendation 2.3 to ensure that the walkdowns address the integrated effects of severe storms as well as seismic and flooding events, in light of the ACRS recommendations. This expansion of NTTF Recommendation 2.3 will have no net impact on the proposed staff resources stated in SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lesson Learned," dated October 3, 2011.
• ACRS Recommendation 1(c)—"Actions related to NTTF Recommendation 4.1 should be expanded to include issuance of an advanced notice of proposed rulemaking and requiring licensee to provide an assessment of capabilities to cope with an extended station blackout (SBO)." • ACRS Recommendation 2(a)—"Performance-based criteria to mitigate and manage an extended SBO should be considered as an alternative to the specific coping times proposed in Recommendation 4.1."	The NRC staff expanded NTTF Recommendation 4.1 to include an advanced notice of proposed rulemaking (ANPR) and performance-based criteria for an extended SBO, in light of the ACRS recommendations and Commission direction in SRM-SECY-11-0124. This expansion of NTTF Recommendation 4.1 will have no net impact on the proposed staff resources stated in SECY-11-0137. Additionally, the Order associated with NTTF Recommendation 4.2 does include performance-based criteria for SBO coping times.

ACRS Recommendations Incorporated into Tier 1 Activities	
ACRS Recommendation	**Staff Response**
• ACRS Conclusion 1—"Rulemaking activities related to strengthening of SBO mitigation capability should be expedited."	The NRC staff accelerated NTTF Recommendation 4.1 as a result of the Commission's decision in Staff Requirements Memorandum (SRM)-SECY-11-0124, "Recommended Actions To Be Taken Without Delay from the Near-Term Task Force Report," dated October 18, 2011. The staff has designated the SBO rulemaking as a high-priority rulemaking with a completion goal of 24 to 30 months. This acceleration of NTTF Recommendation 4.1 will have no net impact on the proposed staff resources stated in SECY-11-0137.
• ACRS Recommendation 1(d)—"Actions related to NTTF Recommendation 5.1 should also be applied to BWR plants with Mark II containments."	The NRC staff expanded NTTF Recommendation 5.1 to include BWR Mark II containments, in light of the ACRS recommendations. This expansion of NTTF Recommendation 5.1 will have no net impact on the proposed staff resources stated in SECY-11-0137.
• ACRS Recommendation 1(f)—"Information should be requested from licensees regarding current plant-specific spent fuel pool (SFP) instrumentation, power supplies, and sources of makeup and cooling water." • ACRS Conclusion 5—"Staff Tier 1 Recommendation 7.1-2, 'Develop and issue order to licensees to provide reliable SFP instrumentation,' should be reconsidered. Schedules for SFP instrumentation improvements and other modifications to the SFP should be informed by quantification of the contribution made by SFPs to the overall plant risk."	The NRC staff enhanced NTTF Recommendation 7.1 and the associated SFP instrumentation Order in light of the ACRS recommendations. The staff used information gathered from all available resources regarding current plant-specific SFP instrumentation to inform the associated Order. This enhancement of NTTF Recommendation 7.1 will have no net impact on the proposed staff resources stated in SECY-11-0137.

ACRS Recommendations Incorporated into Tier 2 Activities	
ACRS Recommendation	**Staff Response**
• ACRS Recommendation 1(a)—"Actions related to NTTF Recommendation 2.1 should be expanded to include an expedited update of the applicable regulatory guidance, methods, and data for external flooding to ensure that outdated guidance and acceptance criteria are not used in the reevaluations."	The NRC staff will expand its actions related to NTTF Recommendation 2.1 to include "other external hazards" in light of Section 402 of the Consolidated Appropriations Act, 2012 (Public Law 112 74) and the ACRS recommendations. This is a new Tier 2 activity. However, in the Tier 1 actions associated with reevaluating seismic and flooding hazards, licensees will use the present-day regulatory guidance and methodologies that are currently being applied to ongoing reviews of ESP and COL applications.
• ACRS Recommendation 1(f)—"Information should be requested from licensee regarding current plant-specific SFP instrumentation, power supplies, and sources of makeup and cooling water." • ACRS Conclusion 5—"Staff Tier 1 Recommendation 7.1-2, "Develop and issue order to licensees to provide reliable SFP instrumentation," should be reconsidered. Schedules for SFP instrumentation improvements and other modifications to the SFP should be informed by quantification of the contribution made by SFPs to the overall plant risk."	The NRC staff will enhance NTTF Recommendations 7.2–7.5 in light of the ACRS recommendations. The staff will use information gathered from all available resources regarding current plant-specific SFP power supplies, and sources of makeup and cooling water, to inform future actions. These enhancements of NTTF Recommendations 7.2–7.5 will have no net impact on the proposed staff resources stated in SECY-11-0137.

ACRS Recommendations Incorporated into Tier 3 Activities[1]	
ACRS Recommendation	**Staff Response**
• ACRS Recommendation 2(e)—“Selected reactor and containment instrumentation should be enhanced to withstand beyond-design-basis accident conditions.” • Conclusion 4—“Tier 2 recommendations should be expanded to include the additional actions recommended in our October 13, 2011, report regarding enhancement of selected reactor and containment instrumentation, and the need to proactively engage in efforts to capture and analyze data from the Fukushima event.”	The NRC staff will develop a new action on “reactor and containment instrumentation withstanding beyond-design-basis conditions” and add it to the Tier 3 actions that the NRC will take in response to the Fukushima lessons–learned.
• ACRS Recommendation 1(e)—“Discussions with stakeholders should be initiated regarding near-term actions for additional hydrogen control and mitigation measures in reactor buildings for plants with Mark I and Mark II containments.”	The NRC staff will include discussions with stakeholders in its Tier 3 actions associated with NTTF Recommendation 6.
• ACRS Recommendation 2(b)—“Recommendation 6 should be expanded to include a requirement for BWR plants with Mark I and Mark II containments to implement combustible gas control measures in reactor buildings as a near-term defense-in-depth measure.”	The NRC staff will enhance the Tier 3 actions associated with NTTF Recommendation 6 to include the implementation of combustible gas control measures in reactor buildings.

1 The resource estimates associated with the incorporation of the above ACRS Recommendation into Tier 3 activities will be described in detail in the staff’s 9-month SECY due to the Commission in July 2012.

ACRS Recommendations Incorporated into Tier 3 Activities	
ACRS Recommendation	**Staff Response**
• ACRS Recommendation 2(c)—"Recommendation 6 should be expanded to include an assessment of the vulnerabilities introduced by shared ventilation systems or shared stacks in multi-unit."	The NRC staff will enhance the Tier 3 actions associated with NTTF Recommendation 6 to include vulnerabilities introduced by shared ventilation systems or shared stacks in multiunit sites.
• ACRS Recommendation 1(g)—"Actions related to NTTF Recommendation 8 should be expanded to included fire response procedures." • ACRS Recommendation 2(d)—"Integration of onsite emergency response capabilities envisioned by Recommendation 8 should be expanded to include fire response procedures."	The NRC staff evaluated how to appropriately integrate the fire response procedure into a licensee's onsite emergency response capabilities and determined that the fire response procedures would be best considered with the agency's Tier 3 actions associated with NTTF Recommendation 3.
• ACRS Conclusion 3—"NTTF Recommendation 10.2 regarding evaluation of the command and control structure and qualifications of decision makers should be initiated in parallel with Tier 1 activities related to integration of onsite emergency actions."	The NRC staff evaluated how to appropriately initiate the "evaluation of the command and control structure and qualifications of decision makers" and determined that they would be best considered with the agency's Tier 3 actions associated with NTTF Recommendation 10.

ACRS Recommendations Addressed by Other NRC Processes or Programs	
ACRS Recommendation	**Staff Response**
• ACRS Recommendation 2(f)—"The NRC should proactively engage in efforts to define and participate in programs to capture and analyze data from the Fukushima event to enhance understanding of severe accident phenomena, including BWR melt progressions, seawater addition effects, hydrogen transport and combustion, and safety systems operability."	The NRC staff in the Office of Nuclear Regulatory Research (RES) is currently working on capturing and analyzing Fukushima data to enhance the agency's understanding of severe accident phenomena.
• ACRS Recommendation 3—"Licensing actions requiring the granting of containment accident pressure (CAP) credit should be suspended until the implications of post-Fukushima containment pressure control measures are understood."	The NRC staff determined that CAP credit will continue to be reviewed on a case-by-case basis.

New Tier 2 Activity–NTTF Recommendation 2.1
Other Natural External Hazards Reevaluations

The NTTF recommends the NRC require licensees to reevaluate and upgrade as necessary the design basis of structures, systems, and components (SSCs) important to safety for protection against updated seismic and flooding hazards . The ACRS recommended expanding this recommendation to include other natural external hazards other than seismic and flooding. The Consolidated Appropriations Act, Public Law 112-074, mandates the NRC to require licensees to reevaluate the external hazards at their sites and to require updates to their design basis, if necessary.

Regulations and Guidance

1. General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that SSCs important to safety be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2. GDC 4, "Environmental and Dynamic Effects Design Bases," requires, in part, that SSCs that are important to safety be adequately protected against the effects of missiles resulting from events and conditions outside the plant.

3. GDC 44, "Cooling Water," states, in part, that a system to transfer heat from SSCs important to safety to an ultimate heat sink (UHS) shall be provided. The system safety function shall be to transfer the combined heat load of these SSCs under normal operating and accident conditions.

4. The regulations in Subpart B, "Evaluation Factors for Stationary Power Reactor Site Applications On or After January 10, 1997, " to 10 CFR Part 100, "Reactor Site Criteria," state, in part, that meteorological characteristics of the site that are necessary for safety analysis or that may have an impact upon plant design (such as maximum probable wind speed and precipitation) must be identified and characterized (10 CFR 100.20(c)(2)). The regulations further state, in part, that the physical characteristics of the site, including meteorology, 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)).

5. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" contains the following sections of interest:

    a. Section 2.3.1, "Regional Climatology"
    b. Section 2.4.2, "Floods"
    c. Section 2.4.11, "Low Water Considerations"

d. Section 3.3.1, “Wind Loadings”
e. Section 3.3.2, “Tornado Loadings”
f. Section 3.5.1.4, “Missiles Generated by Tornadoes and Extreme Winds”
g. Section 5.4.7, “Residual Heat Removal (RHR) System”
h. Section 6.2.1, “Containment Functional Design”
i. Section 6.2.2, “Containment Heat Removal Systems”
j. Section 6.4, “Control Room Habitability System”
k. Section 9.1.3, “Spent Fuel Pool Cooling and Cleanup System”
l. Section 9.2.2, “Reactor Auxiliary Cooling Water Systems”

6. Interim Staff Guidance DC/COL-ISG-7, “Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures,” was issued final on October 9, 2009.

7. Regulatory Guide (RG) 1.27, “Ultimate Heat Sink for Nuclear Power Plants,” Revision 2, was issued January 1976.

8. RG 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Revision 1, was issued March 2007.

9. RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” was issued October 2011.

<u>Staff Assessment and Basis for Prioritization</u>

As a follow-on activity to the completion of the Tier 1 actions on seismic and flooding hazards associated with NTTF Recommendation 2.1, the staff concludes that the recommendation should be enhanced to include other natural hazards (e.g., meteorological phenomena) that could affect the safety of power reactors in the U.S. This expansion was suggested to the staff by the ACRS and was subsequently mandated to the NRC in Section 402 of the Consolidated Appropriations Act of 2012.

ACRS letter dated October 13, 2011 (ADAMS Accession No. ML11284A136), recommended that the staff should expand actions related to NTTF Recommendation 2.3 to include:

> The integrated effects of severe storms as well as seismic and flooding events.

The Consolidated Appropriations Act, Public Law 112-074, was signed into law on December 23, 2011. Section 402 clarified the scope of the staff’s reevaluation of licensees’ design bases 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.

Other Natural External Hazards. The NRC will undertake regulatory actions to ensure that SSCs important to safety will withstand other natural external hazards. These other external hazards can be considered to include meteorological phenomena such as wind and missile loads from tornadoes and hurricanes, maximum rainfall rates and snow and ice load for roof design, drought and other low-water conditions that may reduce or limit the available safety-related cooling water supply, extreme maximum and minimum ambient temperatures for normal plant heat sink and containment heat removal systems (post-accident), and meteorological conditions related to the maximum evaporation and drift loss and minimum water cooling for the UHS design. Flooding reevaluations and walkdowns in response to Tier 1 NTTF Recommendations 2.1 and 2.3 will address reevaluation of flood hazards for each flood causing mechanism, based on present-day methodologies and regulatory guidance. This will include analyses 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.

The staff's assessment of the expansion of NTTF Recommendation 2.1 indicates that plants may differ in the way they protect against natural phenomena. The staff concluded that sufficient regulatory guidance currently exists to permit licensee reevaluations. However, the staff noted that results of inspections of SSCs at Fukushima Dai-ichi and Dai-ni Nuclear Power Stations may help inform the implementation of this recommendation. To the extent practical, the new information on the events at Fukushima Dai-ichi and Dai-ni should be incorporated into the reevaluations.

The staff concludes that this recommendation would improve safety. However, the staff also noted that the implementation of this recommendation would require significant resources for both licensees and the NRC, as well as specialized expertise to review licensee reevaluations and to document results of staff evaluations. Since sufficient resource flexibility, including availability of critical skill sets, does not exist at this time, the staff prioritized this action as a Tier 2 recommendation. Albeit very low, seismic and flooding hazards are expected to be the dominant risks to the operating fleet of plants from external hazards and therefore have been given priority as Tier 1 activities.

<u>Staff Actions</u>

Once sufficient expertise and resources are available, the NRC staff plans to undertake regulatory activities to do the following:

1. Continue stakeholder interactions to discuss the technical basis and acceptance criteria for conducting a reevaluation of site-specific external natural hazards. These interactions will also help to define guidelines for the application of current regulatory guidance and methodologies being used for early site permit and combined license reviews to the reevaluation of hazards at operating reactors.

2. Develop and issue a request for information to licensees pursuant to 10 CFR 50.54(f) to (1) reevaluate site-specific external natural hazards using the methodology discussed in Item 1 above, and (2) identify actions that have been taken, or are planned, to address plant-specific issues associated with the updated natural external hazards (including potential changes to the licensing or design basis of a plant).

3. Evaluate licensee responses and take appropriate regulatory action to resolve issues associated with updated site-specific natural external hazards.

Unique Implementation Challenges

The staff recognizes that the NRC and industry have limited, specialized expertise (e.g., physical scientists, hydrologists) to complete the actions associated with this recommendation.

Schedules and Milestones

Reevaluation of Other Natural External Hazards:

I. Issue a 10 CFR 50.54(f) letter 6 months following initiation of action.

   a. Initiate stakeholder interaction and technical development (e.g., methods, technical basis, acceptance criteria).

   b. Develop a 10 CFR 50.54(f) letter.

   c. Issue a 10 CFR 50.54(f) letter.

II. Evaluate licensee responses to the 10 CFR 50.54(f) letter, based on a timeline to be developed during stakeholder interactions, taking into account available resources.

   a. Write a safety evaluation or NUREG to document staff conclusions.

III. Issue orders to licensees (if needed), 3 months following a decision to issue orders.

   a. Develop the regulatory basis and draft orders.

   b. Issue orders.

IV. Initiate inspection activities, on a schedule to be determined

   a. Develop temporary instructions.

   b. Conduct inspections and document results.

V. Issue letters to close out the 10 CFR 50.54(f) letter and orders, 1 month after last inspection.

Resources for Other Natural External Hazards Reevaluations

Activity	Resource Category	Specific Expertise Needed	Estimated FTE	Locations of Most Applicable Expertise within NRC
I. Develop 10 CFR 50.54(f) letter	Project/Program Management	Plant Licensing	0.3	NRR
	Technical	Physical Science	0.3	NRO, NRR
		Hydrology	0.2	NRO, NRR
		Electrical Engineering; Structural Engineering; Plant Systems	0.1	NRR, NRO
	Legal	Plant Licensing	0.1	OGC
II. Evaluate licensee responses to 10 CFR 50.54(f) letter	Project/Program Management	Plant Licensing	0.3	NRR
	Technical	Physical Science	3.8	NRO, NRR
		Hydrology	1.4	NRO, NRR
		Electrical Engineering; Structural Engineering; Plant Systems	3.0	NRR, NRO
	Legal	Plant Licensing	0.2	OGC
III. Issue orders to licensees (if needed)	Project/Program Management	Plant Licensing	0.3	NRR
	Legal	Plant Licensing	0.2	OGC
	Technical	Hydrology	0.1	NRO, NRR
		Electrical Engineering; Structural Engineering; Plant Systems	0.3	NRR, NRO
IV. Conduct inspection activities	Regional Inspection	Inspection	1.0	All Regions
	Project/Program Management	Inspection Program Management	0.3	NRR
	Technical	Hydrology	0.1	NRO, NRR
		Electrical Engineering; Structural Engineering; Plant Systems	0.3	NRR, NRO

V. Close out 10 CFR 50.54(f) letter and orders	Project/Program Management	Project Management	0.3	NRR
	Legal	Plant Licensing	0.2	OGC
Total FTE			**12.8**	

7590-01-P

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
ALL POWER REACTOR LICENSEES AND HOLDERS OF CONSTRUCTION PERMITS IN ACTIVE OR DEFERRED STATUS	)	Docket Nos. (as shown in Attachment 1) License Nos. (as shown in Attachment 1) or Construction Permit Nos. (as shown in Attachment 1))
	)	
	)	EA-12-XXX

ORDER MODIFYING LICENSES WITH REGARD TO REQUIREMENTS FOR MITIGATION STRATEGIES FOR BEYOND-DESIGN-BASIS EXTERNAL EVENTS (EFFECTIVE IMMEDIATELY)

I.

The Licensees and construction permits (CP) holders[1] identified in Attachment 1 to this Order hold licenses and CPs issued by the U.S. Nuclear Regulatory Commission (NRC or Commission) authorizing operation and/or construction of nuclear power plants in accordance with the Atomic Energy Act of 1954, as amended, and Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

II.

On March 11, 2011, a magnitude 9.0 earthquake struck off the coast of the Japanese island of Honshu. The earthquake resulted in a large tsunami, estimated to have exceeded 14 meters (45 feet) in height, that inundated the Fukushima Dai-ichi Nuclear Power Plant site.

1 *CP holders, as used in this Order, includes CPs, in active or deferred status, as identified in Attachment 1 to this Order (i.e., Watts Bar, Unit 2; and Bellefonte, Units 1 and 2)*

The earthquake and tsunami produced widespread devastation across northeastern Japan and significantly affected the infrastructure and industry in the northeastern coastal areas of Japan.

When the earthquake occurred, Fukushima Dai-ichi Units 1, 2, and 3 were in operation and Units 4, 5, and 6 were shut down for routine refueling and maintenance activities. The Unit 4 reactor fuel was offloaded to the Unit 4 spent fuel pool (SFP). Following the earthquake, the three operating units automatically shut down and offsite power was lost to the entire facility. The emergency diesel generators (EDGs) started at all six units providing alternating current (ac) electrical power to critical systems at each unit. The facility response to the earthquake appears to have been normal.

Approximately 40 minutes following the earthquake and shutdown of the operating units, the first large tsunami wave inundated the site, followed by additional waves. The tsunami caused extensive damage to site facilities and resulted in a complete loss of all ac electrical power at Units 1 through 5, a condition known as station blackout. In addition, all direct current electrical power was lost early in the event on Units 1 and 2 and for some period of time at the other units. Unit 6 retained the function of one air-cooled EDG. Despite their actions, the operators lost the ability to cool the fuel in the Unit 1 reactor after several hours, in the Unit 2 reactor after about 70 hours, and in the Unit 3 reactor after about 36 hours, resulting in damage to the nuclear fuel shortly after the loss of cooling capabilities.

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

As directed by the Commission's Staff Requirement Memorandum (SRM) for SECY-11-0093, the NRC staff reviewed the NTTF recommendations within the context of the NRC's existing regulatory framework and considered the various regulatory vehicles available to the NRC to implement the recommendations. SECY-11-0124 and SECY-11-0137 established the staff's prioritization of the recommendations based upon the potential safety enhancements.

Since receiving the Commission's direction in SRM-SECY-11-0124 and SRM-SECY-11-0137, the NRC staff conducted public meetings to discuss enhanced mitigation strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities following beyond-design-basis external events. At these meetings, the industry described its proposal for a Diverse and Flexible Mitigation Capability (FLEX), as documented in the Nuclear Energy Institute's (NEI's) letter, dated December 16, 2011, letter (Agency Documents Access and Management System (ADAMS) Accession No. ML11353A008). FLEX is proposed as a strategy to fulfill the key safety functions of core cooling, containment integrity, and spent fuel cooling. Stakeholder input influenced the staff to pursue a more performance-based approach to improve the safety of operating power reactors than envisioned in NTTF Recommendation 4.2, SECY-11-0124, and SECY-11-0137.

Current regulatory requirements and existing plant capabilities allow the NRC to conclude that a sequence of events such as the Fukushima Dai-ichi accident is unlikely to occur in the U.S. Therefore, continued operation and continued licensing activities do not pose an imminent threat to public health and safety. However, NRC's assessment of new insights from the events at

Fukushima Dai-ichi leads the staff to conclude that additional requirements must be imposed on Licensees or CP holders to increase the capability of nuclear power plants to mitigate beyond-design-basis external events. These additional requirements are needed to provide adequate protection to public health and safety, as set forth in Section III of this Order.

Guidance and strategies required by this Order would be available if the loss of power, motive force and normal access to the ultimate heat sink to prevent fuel damage in the reactor and SFP affected all units at a site simultaneously. This Order requires a three-phase approach for mitigating beyond-design-basis external events. The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from off site. The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely.

Additional details on an acceptable approach for complying with this Order will be contained in final Interim Staff Guidance (ISG) scheduled to be issued by the NRC in August 2012. This guidance will also include a template to be used for the plan that will be submitted in accordance with Section IV, Condition C.1 below.

III.

Reasonable assurance of adequate protection of the public health and safety and assurance of the common defense and security are the fundamental NRC regulatory objectives. Compliance with NRC requirements plays a critical role in giving the NRC confidence that Licensees or CP holders are maintaining an adequate level of public health and safety and common defense and security. While compliance with NRC requirements presumptively assures adequate protection, new information may reveal that additional requirements are warranted. In such situations, the Commission may act in accordance with its statutory authority under Section 161 of the Atomic Energy Act of 1954, as amended, to require Licensees or CP holders to take action in order to protect health and safety and common defense and security.

To protect public health and safety from the inadvertent release of radioactive materials, the NRC's defense-in-depth strategy includes multiple layers of protection: (1) prevention of accidents by virtue of the design, construction, and operation of the plant; (2) mitigation features to prevent radioactive releases should an accident occur; and (3) emergency preparedness programs that include measures such as sheltering and evacuation. The defense-in-depth strategy also provides for multiple physical barriers to contain the radioactive materials in the event of an accident. The barriers are the fuel cladding, the reactor coolant pressure boundary, and the containment. These defense-in-depth features are embodied in the existing regulatory requirements and thereby provide adequate protection of the public health and safety.

Following the events of September 11, 2001, the NRC issued Order EA-02-026, dated February 25, 2002, which required Licensees to develop mitigating strategies related to the key safety functions of core cooling, containment, and SFP cooling. NEI Document 06-12, "B.5.b Phase 2 & 3 Submittal Guideline" (ADAMS Accession No. ML070090060) provides guidelines that describe the necessary mitigating strategies. The NRC endorsed these guidelines in a letter dated December 22, 2006, designated as Official Use Only. Those mitigating strategies were developed in the context of a localized event that was envisioned to challenge portions of a single unit. The events at Fukushima, however, demonstrate that beyond-design-basis external events may adversely affect: (i) more than one unit at a site with two or more units, and (ii) multiple safety functions at each of several units located on the same site.

The events at Fukushima further highlight the possibility that extreme natural phenomena could challenge the prevention, mitigation, and emergency preparedness defense-in-depth layers. To address the uncertainties associated with beyond-design-basis external events, the

NRC is requiring additional defense-in-depth measures at licensed nuclear power reactors so that the NRC can continue to have reasonable assurance of adequate protection of public health and safety in mitigating the consequences of a beyond-design-basis external event.

The strategies and guidance developed and implemented by Licensees or CP holders in response to the requirements imposed by this Order will provide the necessary capabilities to supplement those of the permanently installed plant structures, systems, and components that could become unavailable following beyond-design-basis external events. These strategies and guidance will enhance the safety and preparedness capabilities established following September 11, 2001, and codified as 10 CFR 50.54(hh)(2). In order to address the potential for more widespread effects of beyond design basis external events, this Order requires strategies with increased capacity to implement protective actions concurrently at multiple units at a site. The strategies shall be developed to add multiple ways to maintain or restore core cooling, containment and SFP cooling capabilities in order to improve the defense-in-depth of licensed nuclear power reactors.

Accordingly, the NRC has concluded that there is a need to redefine the level of protection of public health and safety regarded as adequate under the provisions of the backfit rule, 10 CFR 50.109(a)(4)(iii), and is requiring Licensee or CP holder action to meet that new level of protection. In addition, pursuant to 10 CFR 2.202, the NRC finds that the public health, safety and interest require that this Order be made immediately effective.

The Commission has determined that adequate protection of public health and safety requires that power reactor Licensees and CP holders develop, implement and maintain guidance and strategies to restore or maintain core cooling, containment, and SFP cooling capabilities in the event of a beyond-design-basis external event. These new requirements provide a greater mitigation capability consistent with the overall defense-in-depth philosophy, and, therefore, greater assurance that the challenges posed by beyond-design-basis external events to power reactors do not pose an undue risk to public health and safety. In order to provide reasonable assurance of adequate protection of public health and safety, all operating reactor licenses and CPs under Part 50 identified in Attachment 1 to this Order shall be modified to include the requirements identified in Attachment 2 to this Order. All combined licenses (COLs) under Part 52 identified in Attachment 1 to this Order shall be modified to include the requirements identified in Attachment 3 to this Order.

IV.

Accordingly, pursuant to Sections 161b, 161i, 161o, and 182 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.202, and 10 CFR Parts 50 and 52, IT IS HEREBY ORDERED, EFFECTIVE IMMEDIATELY, THAT ALL LICENSES AND CONSTRUCTION PERMITS IDENTIFIED IN ATTACHMENT 1 TO THIS ORDER ARE MODIFIED AS FOLLOWS:

A. 1. All holders of CPs issued under Part 50 shall, notwithstanding the provisions of any Commission regulation or CPs to the contrary, comply with the requirements described in Attachment 2 to this Order except to the extent that a more stringent requirement is set forth in the CP. These CP holders shall complete full implementation **prior to issuance of an operating license.**

2. All holders of operating licenses issued under Part 50 shall, notwithstanding the provisions of any Commission regulation or license to the contrary, comply with the requirements described in Attachment 2 to this Order except to the extent that a more stringent requirement is set forth in the license. These Licensees shall promptly start implementation of the requirements in Attachment 2 to the Order and shall complete full implementation **no later than two (2) refueling cycles**

after submittal of the overall integrated plan, as required in Condition C.1.a, or December 31, 2016, whichever comes first.

3. All holders of COLs issued under Part 52 shall, notwithstanding the provisions of any Commission regulation or license to the contrary, comply with the requirements described in Attachment 3 to this Order except to the extent that a more stringent requirement is set forth in the license. These Licensees shall promptly start implementation of the requirements in Attachment 3 to the Order and shall complete full implementation prior to initial fuel load.

B. 1. All Licensees and CP holders shall, within **twenty (20) days** of the date of this Order, notify the Commission, (1) if they are unable to comply with any of the requirements described in Attachment 2 or Attachment 3, (2) if compliance with any of the requirements is unnecessary in their specific circumstances, or (3) if implementation of any of the requirements would cause the Licensee or CP holder to be in violation of the provisions of any Commission regulation or the facility license. The notification shall provide the Licensees' or CP holders' justification for seeking relief from or variation of any specific requirement.

2. Any Licensee or CP holder that considers that implementation of any of the requirements described in Attachment 2 or Attachment 3 to this Order would adversely impact safe and secure operation of the facility must notify the Commission, within **twenty (20) days** of this Order, of the adverse safety impact, the basis for its determination that the requirement has an adverse safety impact, and either a proposal for achieving the same objectives specified in Attachment 2 or Attachment 3 requirement in question, or a schedule for modifying the facility to address the adverse safety condition. If neither approach is appropriate, the Licensee or CP holder must supplement its response to Condition B.1 of this Order

to identify the condition as a requirement with which it cannot comply, with attendant justifications as required in Condition B.1.

C. 1. a. All holders of operating licenses issued under Part 50 shall by **February 28, 2013**, submit to the Commission for review an overall integrated plan including a description of how compliance with the requirements described in Attachment 2 will be achieved.

b. All holders of CPs issued under Part 50 or COLs issued under Part 52 shall, within **one (1) year** after issuance of the final ISG, submit to the Commission for review an overall integrated plan including a description of how compliance with the requirements described in Attachment 2 or Attachment 3 will be achieved.

2. All Licensees and holders of CPs shall provide an initial status report **sixty (60) days** following issuance of the final ISG and at **six (6)-month** intervals following submittal of the overall integrated plan, as required in Condition C.1, which delineates progress made in implementing the requirements of this Order.

3. All Licensees and CP holders shall report to the Commission when full compliance with the requirements described in Attachment 2 or Attachment 3 is achieved.

Licensee or CP holders responses to Conditions B.1, B.2, C.1, C.2, and C.3, above shall be submitted in accordance with 10 CFR 50.4 and 10 CFR 52.3, as applicable.

As applicable, the Director, Office of Nuclear Reactor Regulation or the Director, Office of New Reactors may, in writing, relax or rescind any of the above conditions upon demonstration by the Licensee or CP holder of good cause.

V.

In accordance with 10 CFR 2.202, the Licensee or CP holder must, and any other person adversely affected by this Order may, submit an answer to this Order, and may request a hearing on this Order, **within 20 days** of the date of this Order. Where good cause is shown, consideration will be given to extending the time to answer or to request a hearing. A request for extension of time in which to submit an answer or request a hearing must be made in writing to the Director, Office of Nuclear Reactor Regulation or to the Director, Office of New Reactors, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. The answer may consent to this Order.

If a hearing is requested by a Licensee, CP holder or a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearings. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained. Pursuant to 10 CFR 2.202(c)(2)(i), the licensee, CP holder or any other person adversely affected by this Order, may, in addition to demanding a hearing, at the time the answer is filed or sooner, move the presiding officer to set aside the immediate effectiveness of the Order on the ground that the Order, including the need for immediate effectiveness, is not based on adequate evidence but on mere suspicion, unfounded allegations, or error.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E-Filing rule (72 FR 49139, August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by telephone at (301) 415-1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/apply-certificates.html*. System requirements for accessing the E-Submittal server are detailed in NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at *http://www.nrc.gov/site-help/esubmittals.html*. Participants may attempt to use other software not listed on the web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E-Filing rule, the participant must file the document using the NRC's online, web-based submission form. In order to serve documents through the Electronic Information Exchange, users will be required to install a web browser plug-in from the NRC web site. Further information on the web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public web site at *http://www.nrc.gov/site-help/esubmittals.html*.

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at *http://www.nrc.gov/site-help/e-submittals.html*. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically using the agency's adjudicatory E-Filing system may seek assistance by contacting the NRC Meta System Help Desk through the "Contact Us" link located on the NRC web site at *http://www.nrc.gov/site-help/e-submittals.html*, by e-mail at MSHD.Resource@nrc.gov, or by a toll-free call at (866) 672-7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using EFiling, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket, which is available to the public at *http://ehd.nrc.gov/EHD_Proceeding/home.asp*, unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

If a person other than the Licensee or CP holder requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.309(d).

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above shall be final twenty (20) days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section IV shall be final when the extension expires if a hearing request has not been received. AN ANSWER OR A REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

FOR THE NUCLEAR REGULATORY COMMISSION

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Michael R. Johnson, Director
Office of New Reactors

Dated this ____ day of ____________ 2012

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

Arkansas Nuclear One, Units 1 and 2
Entergy Nuclear Operations, Inc.
London, AR
Docket Nos. 50-313 and 50-368
License Nos. DPR-51 and NPF-6

Beaver Valley Power Station, Units 1 and 2
First Energy Nuclear Operating Co.
Shippingport, PA
Docket Nos. 50-334 and 50-412
License Nos. DPR-66 and NPF-73

Bellefonte Nuclear Power Station, Units 1 and 2
Tennessee Valley Authority
Scottsboro, AL
Docket Nos. 50-438 and 50-439
Construction Permit Nos. CPPR-122 and CPPR-123

Braidwood Station, Units 1 and 2
Exelon Generation Co., LLC
Braceville, IL
Docket Nos. 50-456 and 50-457
License Nos. NPF-72 and NPF-77

Browns Ferry Nuclear Plant, Units 1, 2 and 3
Tennessee Valley Authority
Athens, AL
Docket Nos. 50-259, 50-260, and 50-296
License Nos. DPR-33, DPR-52 and DPR-68

Brunswick Steam Electric Plant, Units 1 and 2
Carolina Power & Light Co.
Southport, NC
Docket Nos. 50-325 and 50-324
License Nos. DPR-71 and DPR-62

Byron Station, Units 1 and 2
Exelon Generation Co., LLC
Byron, IL
Docket Nos. 50-454 and 50-455
License Nos. NPF-37 and NPF-66

Callaway Plant
Union Electric Co.
Fulton, MO
Docket No. 50-483
License No. NPF-30

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Calvert Cliffs Nuclear Power Plant, Inc.
Lusby, MD
Docket Nos. 50-317 and 50-318
License Nos. DPR-53 and DPR-69

Catawba Nuclear Station, Units 1 and 2
Duke Energy Carolinas, LLC
York, SC
Docket Nos. 50-413 and 50-414
License Nos. NPF-35 and NPF-52

Clinton Power Station, Unit 1
Exelon Generation Co., LLC
Clinton, IL
Docket No. 50-461
License No. NPF-62

Columbia Generating Station, Unit 2
Energy Northwest
Richland, WA
Docket No. 50-397
License No. NPF-21

Comanche Peak Steam Electric Station, Units 1 and 2
Luminant Generation Co., LLC
Glen Rose, TX
Docket Nos. 50-445 and 50-446
License Nos. NPF-87 and NPF-89

Cooper Nuclear Station
Nebraska Public Power District
Brownville, NE
Docket No. 50-298
License No. DPR-46

Crystal River Nuclear Generating Plant, Unit 3
Florida Power Corp.
Crystal River, FL
Docket No. 50-302
License No. DPR-72

Davis-Besse Nuclear Power Station, Unit 1
First Energy Nuclear Operating Co.
Oak Harbor, OH
Docket No. 50-346
License No. NPF-3

Diablo Canyon Nuclear Power Plant, Units 1 and 2
Pacific Gas & Electric Co.
Avila Beach, CA
Docket Nos. 50-275 and 50-323
License Nos. DPR-80 and DPR-82

Donald C. Cook Nuclear Power Plant, Units 1 and 2
Indiana Michigan Power Co.
Bridgman, MI
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

Dresden Nuclear Power Station, Units 2 and 3
Exelon Generation Co., LLC
Morris, IL
Docket Nos. 50-237 and 50-249
License Nos. DPR-19 and DPR-25

Duane Arnold Energy Center
FPL Energy Duane Arnold, LLC
Palo, IA
Docket No. 50-331
License No. DPR-49

Edwin I. Hatch Nuclear Plant, Units 1 and 2
Southern Nuclear Operating Co.
Baxley, GA
Docket Nos. 50-321 and Docket No. 50-366
License Nos. DPR-57 and NPF-5

Fermi, Unit 2
The Detroit Edison Co.
Newport, MI
Docket No. 50-341
License No. NPF-43

Fort Calhoun Station, Unit 1
Omaha Public Power District
Fort Calhoun, NE
Docket No. 50-285
License No. DPR-40

Grand Gulf Nuclear Station, Unit 1
Entergy Nuclear Operations, Inc.
Port Gibson, MS
Docket No. 50-416
License No. NPF-29

H. B. Robinson Steam Electric Plant, Unit 2
Carolina Power & Light Co.
Hartsville, SC
Docket No. 50-261
License No. DPR-23

Hope Creek Generating Station, Unit 1
PSEG Nuclear, LLC
Hancocks Bridge, NJ
Docket No. 50-354
License No. NPF-57

Indian Point Nuclear Generating Station, Units 2 and 3
Entergy Nuclear Operations, Inc.
Buchanan, NY
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

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

Joseph M. Farley Nuclear Plant, Units 1 and 2
Southern Nuclear Operating Co.
Columbia, AL
Docket Nos. 50-348 and 50-364
License Nos. NPF-2 and NPF-8

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

LaSalle County Station, Units 1 and 2
Exelon Generation Co., LLC
Marseilles, IL
Docket Nos. 50-373 and 50-374
License Nos. NPF-11 and NPF-18

Limerick Generating Station, Units 1 and 2
Exelon Generation Co., LLC
Limerick, PA
Docket Nos. 50-352 and 50-353
License Nos. NPF-39 and NPF-85

McGuire Nuclear Station, Units 1 and 2
Duke Energy Carolinas, LLC
Huntersville, NC
Docket Nos. 50-369 and 50-370
License Nos. NPF-9 and NPF-17

Millstone Power Station, Units 2 and 3
Dominion Nuclear Connecticut, Inc.
Waterford, CT
Docket Nos. 50-336 and 50-423
License Nos. DPR-65 and NPF-49

Monticello Nuclear Generating Plant, Unit 1
Northern States Power Company
Monticello, MN
Docket No. 50-263
License No. DPR-22

Nine Mile Point Nuclear Station, Units 1 and 2
Nine Mile Point Nuclear Station, LLC
Scriba, NY
Docket Nos. 50-220 and 50-410
License Nos. DPR-63 and NPF-69

North Anna Power Station, Units 1 and 2
Virginia Electric & Power Co.
Louisa, VA
Docket Nos. 50-338 and 50-339
License Nos. NPF-4 and NPF-7

Oconee Nuclear Station, Units 1, 2, and 3
Duke Energy Carolinas, LLC
Seneca, SC
Docket Nos. 50-269, 50-270, and 50-287
License Nos. DPR-38, DPR-47, and DPR-55

Oyster Creek Nuclear Generating Station, Unit 1
Exelon Generation Co., LLC
Forked River, NJ
Docket No. 50-219
License No. DPR-16

Palisades Nuclear Plant
Entergy Nuclear Operations, Inc.
Covert, MI
Docket No. 50-255
License No. DPR-20

Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Arizona Public Service Company
Wintersburg, AZ
Docket Nos. 50-528, 50-529, and 50-530
License Nos. NPF-41, NPF-51 and NPF-74

Peach Bottom Atomic Power Station, Units 2 and 3
Exelon Generation Co., LLC
Delta, PA
Docket Nos. 50-277 and 50-278
License Nos. DPR-44 and DPR-56

Perry Nuclear Power Plant, Unit 1
First Energy Nuclear Operating Co.
Perry, OH
Docket No. 50-440
License No. NPF-58

Pilgrim Nuclear Power Station
Entergy Nuclear Operations, Inc.
Plymouth, MA
Docket No. 50-293
License No. DPR-35

Point Beach Nuclear Plant, Units 1 and 2
FPL Energy Duane Arnold, LLC
Two Rivers, WI
Docket Nos. 50-266 and 50-301
License Nos. DPR-24 and DPR-27

Prairie Island Nuclear Generating Plant, Units 1 and 2
Northern States Power Co. Minnesota
Welch, MN
Docket Nos. 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Quad Cities Nuclear Power Station, Units 1 and 2
Exelon Generation Co., LLC
Morris, IL
Docket Nos. 50-254 and 50-265
License Nos. DPR-29 and DPR-30

River Bend Station, Unit 1
Entergy Nuclear Operations, Inc.
St. Francisville, LA
Docket No. 50-458
License No. NPF-47

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

St. Lucie Plant, Units 1 and 2
Florida Power & Light Co.
Jensen Beach, FL
Docket Nos. 50-335 and 50-389
License Nos. DPR-67 and NPF-16

Salem Nuclear Generating Station, Units 1 and 2
PSEG Nuclear, LLC
Hancocks Bridge, NJ
Docket Nos. 50-272 and 50-311
License Nos. DPR-70 and DPR-75

San Onofre Nuclear Generating Station, Units 2 and 3
Southern California Edison Co.
San Clemente, CA
Docket Nos. 50-361 and 50-362
License Nos. NPF-10 and NPF-15

Seabrook Station, Unit 1
FPL Energy Seabrook, LLC
Seabrook, NH
Docket No. 50-443
License No. NPF-86

Sequoyah Nuclear Plant, Units 1 and 2
Tennessee Valley Authority
Soddy-Daisy, TN
Docket Nos. 50-327 and 50-328
License Nos. DPR-77 and DPR-79

Shearon Harris Nuclear Power Plant, Unit 1
Carolina Power & Light Co.
New Hill, NC
Docket No. 50-400
License No. NPF-63

South Texas Project, Units 1 and 2
STP Nuclear Operating Co.
Bay City, TX
Docket Nos. 50-498 and 50-499
License Nos. NPF-76 and NPF-80

Surry Nuclear Power Station, Units 1 and 2
Virginia Electric & Power Co.
Surry, VA
Docket Nos. 50-280 and 50-281
License Nos. DPR-32 and DPR-37

Susquehanna Steam Electric Station, Units 1 and 2
PPL Susquehanna, LLC
Salem Township, Luzerne Co., PA
Docket Nos. 50-387 and 50-388
License Nos. NPF-22 and NPF-14

Three Mile Island Nuclear Station, Unit 1
Exelon Generation Co., LLC
Middletown, PA
Docket No. 50-289
License No. DPR-50

Turkey Point Nuclear Generating, Units 3 and 4
Florida Power & Light Co.
Homestead, FL
Docket Nos. 50-250 and 50-251
License Nos. DPR-31 and DPR-41

Vermont Yankee Nuclear Power Plant, Unit 1
Entergy Nuclear Operations, Inc.
Vernon, VT
Docket No. 50-271
License No. DPR-28

Virgil C. Summer Nuclear Station, Unit1
South Carolina Electric & Gas Co.
Jenkinsville, SC
Docket No. 50-395
License No. NPF-12

Vogtle Electric Generating Plant, Units 1, 2, 3, and 4
Southern Nuclear Operating Co.
Waynesboro, GA
Docket Nos. 50-424, 50-425, 52-025, and 52-026
License Nos. NPF-68, NPF-81, NPF-91 and NPF-92

Waterford Steam Electric Station, Unit 3
Entergy Nuclear Operations, Inc.
Killona, LA
Docket No. 50-382
License No. NPF-38

Watts Bar Nuclear Plant, Units 1 and 2
Tennessee Valley Authority
Spring City, TN
Docket No. 5000390 and 5000391
License No. NPF-90 and
Construction Permit No. CPPR-92

Wolf Creek Generating Station, Unit 1
Wolf Creek Nuclear Operating Corp.
Burlington, Coffey County, KS
Docket No. 5000482
License No. NPF-42

REQUIREMENTS FOR MITIGATION STRATEGIES FOR BEYOND-DESIGN-BASIS EXTERNAL EVENTS AT OPERATING REACTOR SITES AND CONSTRUCTION PERMIT HOLDERS

This Order requires a three-phase approach for mitigating beyond-design-basis external events. The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment and spent fuel pool (SFP) cooling capabilities. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from off site. The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely.

(1) Licensees or construction permit (CP) holders shall develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment and SFP cooling capabilities following a beyond-design-basis external event.

(2) These strategies must be capable of mitigating a simultaneous loss of all alternating current (ac) power and loss of normal access to the ultimate heat sink and have adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this Order.

(3) Licensees or CP holders must provide reasonable protection for the associated equipment from external events. Such protection must demonstrate that there is adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this Order.

(4) Licensees or CP holders must be capable of implementing the strategies in all modes.

(5) Full compliance shall include procedures, guidance, training, and acquisition, staging, or installing of equipment needed for the strategies.

REQUIREMENTS FOR MITIGATION STRATEGIES FOR BEYOND-DESIGN-BASIS EXTERNAL EVENTS AT COL HOLDER REACTOR SITES (VOGTLE UNITS 3 AND 4)

Attachment 2 to this order for Part 50 licensees requires a phased approach for mitigating beyond-design-basis external events. The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment and spent fuel pool (SFP) cooling capabilities. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from off site. The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely.

The design bases of Vogtle Units 3 and 4 includes passive design features that provide core, containment and SFP cooling capability for 72 hours, without reliance on alternating current (ac) power. These features do not rely on access to any external water sources since the containment vessel and the passive containment cooling system serve as the safety-related ultimate heat sink. The NRC staff reviewed these design features prior to issuance of the combined licenses for these facilities and certification of the AP1000 design referenced therein. The AP1000 design also includes equipment to maintain required safety functions in the long term (beyond 72 hours to 7 days) including capability to replenish water supplies. Connections are provided for generators and pumping equipment that can be brought to the site to back up the installed equipment. The staff concluded in its final safety evaluation report for the AP1000 design that the installed equipment (and alternatively, the use of transportable equipment) is capable of supporting extended operation of the passive safety systems to maintain required safety functions in the long term. As such, this Order requires Vogtle Units 3 and 4 to address the following requirements relative to the final phase.

(1) Licensees shall develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment and SFP cooling capabilities following a beyond-design-basis external event.

(2) These strategies must be capable of mitigating a simultaneous loss of all ac power and loss of normal access to the normal heat sink and have adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this Order.

(3) Licensees must provide reasonable protection for the associated equipment from external events. Such protection must demonstrate that there is adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this Order.

(4) Licensees must be capable of implementing the strategies in all modes.

(5) Full compliance shall include procedures, guidance, training, and acquisition, staging, or installing of equipment needed for the strategies.

7590-01-P

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
ALL OPERATING BOILING WATER	)	Docket Nos. (as shown in Attachment 1)
REACTOR LICENSEES WITH MARK I AND	)	License Nos. (as shown in Attachment 1)
MARK II CONTAINMENTS	)	EA-12-XXX
	)	

ORDER MODIFYING LICENSES WITH REGARD TO RELIABLE HARDENED CONTAINMENT VENTS (EFFECTIVE IMMEDIATELY)

I.

The Licensees identified in Attachment 1 to this Order hold licenses issued by the U.S. Nuclear Regulatory Commission (NRC or Commission) authorizing operation of nuclear power plants in accordance with the Atomic Energy Act of 1954, as amended, and Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." Specifically, these Licensees operate boiling-water reactors (BWRs) with Mark I and Mark II containment designs.

II.

On March 11, 2011, a magnitude 9.0 earthquake struck off the coast of the Japanese island of Honshu. The earthquake resulted in a large tsunami, estimated to have exceeded 14 meters (45 feet) in height, which inundated the Fukushima Dai-ichi Nuclear Power Plant site. The earthquake and tsunami produced widespread devastation across northeastern Japan, and significantly affected the infrastructure and industry in the northeastern coastal areas of Japan.

When the earthquake occurred, Fukushima Dai-ichi Units 1, 2, and 3 were in operation and Units 4, 5, and 6 were shut down for routine refueling and maintenance activities. The Unit 4 reactor fuel was offloaded to the Unit 4 spent fuel pool. Following the earthquake, the three operating units automatically shut down and offsite power was lost to the entire facility. The emergency diesel generators (EDGs) started at all six units providing alternating current (ac) electrical power to critical systems at each unit. The facility response to the earthquake appears to have been normal.

Approximately 40 minutes following the earthquake and shutdown of the operating units, the first large tsunami wave inundated the site, followed by additional waves. The tsunami caused extensive damage to site facilities and resulted in a complete loss of all ac electrical power at Units 1 through 5, a condition known as station blackout (SBO). In addition, all direct current electrical power was lost early in the event on Units 1 and 2, and for some period of time at the other units. Unit 6 retained the function of one air-cooled EDG. Despite their actions, the operators lost the ability to cool the fuel in the Unit 1 reactor after several hours, in the Unit 2 reactor after about 70 hours, and in the Unit 3 reactor after about 36 hours, resulting in damage to the nuclear fuel shortly after the loss of cooling capabilities.

Operators first considered using the facility's hardened vent to control pressure in the containment within an hour following the loss of all ac power at Unit 1. The Emergency Response Center began reviewing accident management procedures and checking containment venting procedures to determine how to open the containment vent valves without power.[1] However, without adequate core and containment cooling, primary containment (drywell) pressure and temperature in Units 1, 2, and 3 substantially exceeded the design values for the containments. When the operators attempted to vent the containments, they were significantly

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[1] See Institute of Nuclear Power Operations (INPO) report "*INPO 11-005, Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station, Revision 0*," issued November 2011, p. 72

challenged opening the hardened wetwell (suppression chamber) vents because of complications from the prolonged SBO, and high radiation fields that impeded access.

At Fukushima Dai-ichi Units 1, 2, 3, and 4, venting the wetwell involved opening motor- and air-operated valves. Similar features are used in many hardened vent systems that were installed in U.S. BWR Mark I containment plants following issuance of Generic Letter (GL) 89-16, "Installation of a Hardened Wetwell Vent." In the prolonged SBO situation that occurred at Fukushima, operator actions were not possible from the control room because of the loss of power, and the loss of pneumatic supply pressure to the air-operated valves. The resultant delay in venting the containment precluded early injection of coolant into the reactor vessel. The lack of coolant, in turn, resulted in extensive core damage, high radiation levels, hydrogen production and containment failure. The leakage of hydrogen gas into the reactor building precipitated explosions in the secondary containment buildings of Units 1, 3, and 4, and the ensuing damage to the facility contributed to the uncontrolled release of radioactive material to the environment.

Fukushima Dai-ichi Units 1, 2, 3, and 4 use the Mark I containment design; however, because Mark II containment designs are only slightly larger in volume than Mark I containment designs and use wetwell pressure suppression, it can reasonably be concluded that a Mark II under similar circumstances would have suffered similar consequences.

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

As directed by the Staff Requirements Memorandum (SRM) for SECY-11-0093, the NRC staff reviewed the NTTF recommendations within the context of the NRC's existing regulatory framework and considered the various regulatory vehicles available to the NRC to implement the recommendations. SECY-11-0124 and SECY-11-0137 established the staff's prioritization of the recommendations based upon the potential safety enhancements.

Current regulatory requirements and existing plant capabilities allow the NRC to conclude that a sequence of events such as the Fukushima Dai-ichi accident is unlikely to occur in the U.S. Therefore, continued operation and continued licensing activities do not pose an imminent threat to public health and safety. However, NRC's assessment of new insights from the events at Fukushima Dai-ichi leads the staff to conclude that additional requirements must be imposed on Licensees to increase the capability of nuclear power plants to mitigate beyond-design-basis external events. These additional requirements are needed to provide adequate protection to public health and safety, as set forth in Section III of this Order.

In SRM-SECY-11-0137, the Commission directed the NRC staff to take certain actions and provided further guidance including directing the staff to consider filtered vents. The staff has determined that there are policy issues that need to be resolved before any regulatory action can be taken to require Licensees to install filtered vents. These policy issues include consideration of severe accident conditions in the design and operation of the vent, the addition of filters to hardened reliable vents, and consideration of vents in areas other than primary containment. However, the NRC has also determined that Licensees should promptly begin the implementation of short-term actions relating to reliable hardened vents and to focus these actions on improvements that will assist in the prevention of core damage. As such, this Order requires Licensees to take the necessary actions to install reliable hardened venting systems in BWR facilities with Mark I and Mark II containments to assist strategies relating to the prevention of core damage. With respect to the policy issues discussed above, the NRC staff plans to submit a Policy Paper to the Commission in July 2012.

Additional details on an acceptable approach for complying with this Order will be contained in final Interim Staff Guidance (ISG) scheduled to be issued by the NRC in August 2012. This guidance will also include a template to be used for the plan that will be submitted in accordance with Section IV, C.1 below.

III.

Reasonable assurance of adequate protection of the public health and safety and assurance of the common defense and security are the fundamental NRC regulatory objectives. Compliance with NRC requirements plays a critical role in giving the NRC confidence that Licensees are maintaining an adequate level of public health and safety and common defense and security. While compliance with NRC requirements presumptively assures adequate protection, new information may reveal that additional requirements are warranted. In such situations, the Commission may act in accordance with its statutory authority under Section 161 of the Atomic Energy Act of 1954, as amended, to require Licensees to take action in order to protect health and safety and common defense and security.

To protect public health and safety from the inadvertent release of radioactive materials, the NRC's defense-in-depth strategy includes multiple layers of protection: (1) prevention of accidents by virtue of the design, construction and operation of the plant, (2) mitigation features to prevent radioactive releases should an accident occur, and (3) emergency preparedness programs that include measures such as sheltering and evacuation. The defense-in-depth strategy also provides for multiple physical barriers to contain the radioactive materials in the event of an accident. The barriers are the fuel cladding, the reactor coolant pressure boundary, and the containment. These defense-in-depth features are embodied in the existing regulatory requirements and thereby provide adequate protection of public health and safety.

The events at Fukushima Dai-ichi highlight the possibility that extreme natural phenomena could challenge the prevention, mitigation and emergency preparedness defense-in-depth layers. At Fukushima, limitations in time and unpredictable conditions associated with the accident significantly challenged attempts by the responders to preclude core damage and containment failure. In particular, the operators were unable to successfully operate the containment venting system. The inability to reduce containment pressure inhibited efforts to cool the reactor core. If additional backup or alternate sources of power had been available to operate the containment venting system remotely, or if certain valves had been more accessible for manual operation, the operators at Fukushima may have been able to depressurize the containment earlier. This, in turn, could have allowed operators to implement strategies using low-pressure water sources that may have limited damage to the reactor core. Thus, the events at Fukushima demonstrate that reliable hardened vents at BWR facilities with Mark I and Mark II containment designs are important to maintain core and containment cooling.

Accordingly, the NRC has concluded that there is a need to redefine the level of protection of public health and safety regarded as adequate under the provisions of the backfit rule, 10 CFR 50.109(a)(4)(iii), and is requiring Licensee actions to meet the new level of protection. In addition, pursuant to 10 C.F.R. 2.202, the NRC finds that the public health, safety and interest require that this Order be made immediately effective.

The Commission has determined that adequate protection of public health and safety requires that all operating BWR facilities with Mark I and Mark II containments have a reliable hardened venting capability for events that can lead to core damage. These new requirements provide greater mitigation capability consistent with the overall defense-in-depth philosophy, and therefore greater assurance that the challenges posed by severe external events to power reactors do not pose an undue risk to public health and safety. To provide reasonable assurance of adequate protection of public health and safety, all licenses identified in Attachment 1 to this Order shall be modified to include the requirements identified in Attachment 2 to this Order.

IV.

Accordingly, pursuant to Sections 161b, 161i, 161o, and 182 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 C.F.R. § 2.202, “Orders,” and 10 C.F.R. Part 50, IT IS HEREBY ORDERED, EFFECTIVE IMMEDIATELY, THAT ALL LICENSES IDENTIFIED IN ATTACHMENT 1 TO THIS ORDER ARE MODIFIED AS FOLLOWS:

A. All Licensees shall, notwithstanding the provisions of any Commission regulation or license to the contrary, comply with the requirements described in Attachment 2 to this Order except to the extent that a more stringent requirement is set forth in the license. These Licensees shall promptly start implementation of the requirements in Attachment 2 to the Order and shall complete full implementation **no later than two (2) refueling cycles following the submittal of the overall integrated plan, as required in Condition C.1. (schedule to be issued in August 2012), or December 31, 2016, whichever comes first**.

B. 1. All Licensees shall, within **twenty (20) days** of the date of this Order, notify the Commission (1) if they are unable to comply with any of the requirements described in Attachment 2, (2) if compliance with any of the requirements is unnecessary in their specific circumstances, or (3) if implementation of any of the requirements would cause the Licensee to be in violation of the provisions of any Commission regulation or the facility license. The notification shall provide the Licensees' justification for seeking relief from or variation of any specific requirement.

2. Any Licensee that considers that implementation of any of the requirements described in Attachment 2 to this Order would adversely affect the safe and secure operation of the facility must notify the Commission, within **twenty (20) days** of this Order, of the adverse safety impact, the basis for its determination that the requirement has an adverse safety impact, and either a proposal for achieving the same objectives specified in the Attachment 2 requirement in question, or a schedule for modifying the facility to address the adverse safety condition. If neither approach is appropriate, the Licensee must supplement its response to Condition B.1 of this Order to identify the condition as a requirement with which it cannot comply, with attendant justifications as required in Condition B.1.

C. 1. All Licensees shall, **by February 28, 2013**, submit to the Commission for review an overall integrated plan including a description of how compliance with the requirements described in Attachment 2 will be achieved.

2. All Licensees shall provide an initial status report **sixty (60) days** following issuance of the final ISG, and at **six (6)-month intervals** following submittal of the

overall integrated plan, as required in Condition C.1, which delineates progress made in implementing the requirements of this Order.

3. All Licensees shall report to the Commission when full compliance with the requirements described in Attachment 2 is achieved.

Licensee responses to Conditions B.1, B.2, C.1, C.2, and C.3 above shall be submitted in accordance with 10 C.F.R. § 50.4, “Written Communications.”

The Director, Office of Nuclear Reactor Regulation may, in writing, relax or rescind any of the above conditions upon demonstration by the Licensee of good cause.

V.

In accordance with 10 C.F.R. § 2.202, the Licensee must, and any other person adversely affected by this Order may, submit an answer to this Order, and may request a hearing on this Order, within twenty (20) days of the date of this Order. Where good cause is shown, consideration will be given to extending the time to answer or to request a hearing. A request for extension of time in which to submit an answer or request a hearing must be made in writing to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. The answer may consent to this Order.

If a hearing is requested by a Licensee or a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearings. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained. Pursuant to 10 CFR 2.202(c)(2)(i), the licensee or any other person adversely affected by this Order, may, in addition to demanding a hearing, at the time the answer is filed or sooner, move the presiding officer to set aside the immediate effectiveness of the Order on the ground that the Order, including the need for immediate effectiveness, is not based on adequate evidence but on mere suspicion, unfounded allegations, or error.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E-Filing rule (72 FR 49139, August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by telephone at (301) 415-1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/apply-certificates.html*. System requirements for accessing the E-Submittal server are detailed in NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at *http://www.nrc.gov/site-help/esubmittals.html*. Participants may attempt to use other software not listed on the web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E-Filing rule, the participant must file the document using the NRC's online, web-based submission form. In order to serve documents through the Electronic Information Exchange, users will be required to install a web browser plug-in from the NRC web site. Further information on the web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public web site at *http://www.nrc.gov/site-help/esubmittals.html*.

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at *http://www.nrc.gov/site-help/e-submittals.html*. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically using the agency's adjudicatory E-Filing system may seek assistance by contacting the NRC Meta System Help Desk through the "Contact Us" link located on the NRC web site at *http://www.nrc.gov/site-help/e-submittals.html*, by e-mail at MSHD.Resource@nrc.gov, or by a toll-free call at (866) 672-7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using EFiling, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket, which is available to the public at *http://ehd.nrc.gov/EHD_Proceeding/home.asp*, unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

If a person other than the Licensee requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.309(d).

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above shall be final twenty (20) days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section IV shall be final when the extension expires if a hearing request has not been received. AN ANSWER OR A REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

FOR THE NUCLEAR REGULATORY COMMISSION

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Dated this ____ day of March 2012

OPERATING BOILING WATER REACTOR LICENSES
WITH MARK I AND MARK II CONTAINMENTS

Browns Ferry Nuclear Plant, Unit 1 Tennessee Valley Authority Athens, AL Docket No. 50-259 License No. DPR-33	BWR-Mark I
Browns Ferry Nuclear Plant, Unit 2 Tennessee Valley Authority Athens, AL Docket No. 50-260 License No. DPR-52	BWR-Mark I
Browns Ferry Nuclear Plant, Unit 3 Tennessee Valley Authority Athens, AL Docket No. 50-296 License No. DPR-68	BWR-Mark I
Brunswick Steam Electric Plant, Unit 1 Carolina Power and Light Southport, NC Docket No. 50-325 License No. DPR-71	BWR-Mark I
Brunswick Steam Electric Plant, Unit 2 Carolina Power and Light Southport, NC Docket No. 50-324 License No. DPR-62	BWR-Mark I
Columbia Generating Station, Unit 2 Energy Northwest Richland, WA Docket No. 50-397 License No. NPF-21	BWR-Mark II
Cooper Nuclear Station Nebraska Public Power District Brownville, NE Docket No. 50-298 License No. DPR-46	BWR-Mark I

Dresden Nuclear Power Station, Unit 2 Exelon Generation Co., LLC Morris, IL Docket No. 50-237 License No. DPR-19	BWR-Mark I
Dresden Nuclear Power Station, Unit 3 Exelon Generation Co., LLC Morris, IL Docket No. 50-249 License No. DPR-25	BWR-Mark I
Duane Arnold Energy Center FPL Energy Duane Arnold, LLC Palo, IA Docket No. 50-331 License No. DPR-49	BWR-Mark I
Edwin I. Hatch Nuclear Plant, Unit 1 Southern Nuclear Operating Co. Baxley , GA Docket No. 50-321 License No. DPR-57	BWR-Mark I
Edwin I. Hatch Nuclear Plant, Unit 2 Southern Nuclear Operating Co. Baxley , GA Docket No. 50-366 License No. NPF-5	BWR-Mark I
Fermi, Unit 2 The Detroit Edison Co. Newport, MI Docket No. 50-341 License No. NPF-43	BWR-Mark I
Hope Creek Generating Station, Unit 1 PSEG Nuclear, LLC Hancock Bridge, NJ Docket No. 50-354 License No. NPF-57	BWR-Mark I
James A. FitzPatrick Nuclear Power Plant Entergy Nuclear Operations, Inc. Scriba, NY Docket No. 50-333 License No. DPR-59	BWR-Mark I

LaSalle County Station, Unit 1 Exelon Generation Co., LLC Marseilles, IL Docket No. 50-373 License No. NPF-11	BWR-Mark II
LaSalle County Station, Unit 2 Exelon Generation Co., LLC Marseilles, IL Docket No. 50-374 License No. NPF-18	BWR-Mark II
Limerick Generating Station, Unit 1 Exelon Generation Co., LLC Limerick, PA Docket No. 50-352 License No. NPF-39	BWR-Mark II
Limerick Generating Station, Unit 2 Exelon Generation Co., LLC Limerick, PA Docket No. 50-353 License No. NPF-85	BWR-Mark II
Monticello Nuclear Generating Plant, Unit 1 Northern States Power Company Monticello, MN Docket No. 50-263 License No. DPR-22	BWR-Mark I
Nine Mile Point Nuclear Station, Unit 1 Nine Mile Point Nuclear Station, LLC Scriba, NY Docket No. 50-220 License No. DPR-63	BWR-Mark I
Nine Mile Point Nuclear Station, Unit 2 Nine Mile Point Nuclear Station, LLC Scriba, NY Docket No. 50-410 License No. NPF-69	BWR-Mark II
Oyster Creek Nuclear Generating Station, Unit 1 Exelon Generation Co., LLC Forked River, NJ Docket No. 50-219 License No. DPR-16	BWR-Mark I

Peach Bottom Atomic Power Station, Unit 2 Exelon Generation Co., LLC Delta, PA Docket No. 50-277 License No. DPR-44	BWR-Mark I
Peach Bottom Atomic Power Station, Unit 3 Exelon Generation Co., LLC Delta, PA Docket No. 50-278 License No. DPR-56	BWR-Mark I
Pilgrim Nuclear Power Station Entergy Nuclear Operations, Inc. Plymouth, MA Docket No. 50-293 License No. DPR-35	BWR-Mark I
Quad Cities Nuclear Power Station, Unit 1 Exelon Generation Co., LLC Cordova, IL Docket No. 50-254 License No. DPR-29	BWR-Mark I
Quad Cities Nuclear Power Station, Unit 2 Exelon Generation Co., LLC Cordova, IL Docket No. 50-265 License No. DPR-30	BWR-Mark I
Susquehanna Steam Electric Station, Unit 1 PPL Susquehanna, LLC Salem Township, Luzerne Co., PA Docket No. 50-388 License No. NPF-22	BWR-Mark II
Susquehanna Steam Electric Station, Unit 2 PPL Susquehanna, LLC Salem Township, Luzerne Co., PA Docket No. 50-387 License No. NPF-14	BWR-Mark II
Vermont Yankee Nuclear Power Plant, Unit 1 Entergy Nuclear Operations, Inc. Vernon, VT Docket No. 50-271 License No. DPR-28	BWR-Mark I

REQUIREMENTS FOR RELIABLE HARDENED VENT SYSTEMS
AT BOILING-WATER REACTOR FACILITIES WITH
MARK I AND MARK II CONTAINMENTS

1. Hardened Containment Venting System (HCVS) Functional Requirements

Boiling-Water Reactor (BWR) Mark I and Mark II containments shall have a reliable hardened vent to remove decay heat and maintain control of containment pressure within acceptable limits following events that result in the loss of active containment heat removal capability or prolonged Station Blackout (SBO). The hardened vent system shall be accessible and operable under a range of plant conditions, including a prolonged SBO and inadequate containment cooling.

1.1 The design of the HCVS shall consider the following performance objectives:

1.1.1 The HCVS shall be designed to minimize the reliance on operator actions.

1.1.2 The HCVS shall be designed to minimize plant operators' exposure to occupational hazards, such as extreme heat stress, while operating the HCVS system.

1.1.3 The HCVS shall also be designed to minimize radiological consequences that would impede personnel actions needed for event response.

1.2 The HCVS shall include the following design features:

1.2.1 The HCVS shall have the capacity to vent the steam/energy equivalent of 1 percent of licensed/rated thermal power (unless a lower value is justified by analyses), and be able to maintain containment pressure below the primary containment design pressure.

1.2.2 The HCVS shall be accessible to plant operators and be capable of remote operation and control, or manual operation, during sustained operations.

1.2.3 The HCVS shall include a means to prevent inadvertent actuation.

1.2.4 The HCVS shall include a means to monitor the status of the vent system (e.g., valve position indication) from the control room or other location(s). The monitoring system shall be designed for sustained operation during a prolonged SBO.

1.2.5 The HCVS shall include a means to monitor the effluent discharge for radioactivity that may be released from operation of the HCVS. The monitoring system shall provide indication in the control room or other location(s), and shall be designed for sustained operation during a prolonged SBO.

1.2.6 The HCVS shall include design features to minimize unintended cross flow of vented fluids within a unit and between units on the site.

1.2.7 The HCVS shall include features and provision for the operation, testing, inspection and maintenance adequate to ensure that reliable function and capability are maintained.

1.2.8 The HCVS shall be designed for pressures that are consistent with maximum containment design pressures as well as dynamic loading resulting from system actuation.

1.2.9 The HCVS shall discharge the effluent to a release point above main plant structures.

2. <u>Hardened Containment Venting System Quality Standards</u>

The following quality standards are necessary to fulfill the requirements for a reliable HCVS:

2.1 The HCVS vent path up to and including the second containment isolation barrier shall be designed consistent with the design basis of the plant. These items include piping, piping supports, containment isolation valves, containment isolation valve actuators and containment isolation valve position indication components.

2.2 All other HCVS components shall be designed for reliable and rugged performance that is capable of ensuring HCVS functionality following a seismic event. These items include electrical power supply, valve actuator pneumatic supply and instrumentation (local and remote) components.

3. <u>Hardened Containment Venting System Programmatic Requirements</u>

3.1 The Licensee shall develop, implement, and maintain procedures necessary for the safe operation of the HCVS. Procedures shall be established for system operations when normal and backup power is available, and during SBO conditions.

3.2 The Licensee shall train appropriate personnel in the use of the HCVS. The training curricula shall include system operations when normal and backup power is available, and during SBO conditions.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
ALL POWER REACTOR LICENSEES AND HOLDERS OF CONSTRUCTION PERMITS IN ACTIVE OR DEFERRED STATUS	)	Docket Nos. (as shown in Attachment 1) License Nos. (as shown in Attachment 1) or Construction Permit Nos. (as shown in Attachment 1)
	)	
	)	EA-12-XXX

ORDER MODIFYING LICENSES WITH REGARD TO RELIABLE SPENT FUEL POOL INSTRUMENTATION (EFFECTIVE IMMEDIATELY)

I.

The Licensees and construction permit (CP) holders[1] identified in Attachment 1 to this Order hold licenses issued by the U.S. Nuclear Regulatory Commission (NRC or Commission) authorizing operation and/or construction of nuclear power plants in accordance with the Atomic Energy Act of 1954, as amended, and Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

II.

On March 11, 2011, a magnitude 9.0 earthquake struck off the coast of the Japanese island of Honshu. The earthquake resulted in a large tsunami, estimated to have exceeded 14 meters (45 feet) in height, that inundated the Fukushima Dai-ichi Nuclear Power Plant site.

[1] CP holders, as used in this Order, includes CPs, in active or deferred status, as identified in Attachment 1 to this Order (i.e., Watts Bar, Unit 2; and Bellefonte, Units 1 and 2)

The earthquake and tsunami produced widespread devastation across northeastern Japan and significantly affected the infrastructure and industry in the northeastern coastal areas of Japan.

When the earthquake occurred, Fukushima Dai-ichi Units 1, 2, and 3 were in operation and Units 4, 5, and 6 were shut down for routine refueling and maintenance activities. The Unit 4 reactor fuel was offloaded to the Unit 4 spent fuel pool. Following the earthquake, the three operating units automatically shut down and offsite power was lost to the entire facility. The emergency diesel generators (EDGs) started at all six units providing alternating current (ac) electrical power to critical systems at each unit. The facility response to the earthquake appears to have been normal.

Approximately 40 minutes following the earthquake and shutdown of the operating units, the first large tsunami wave inundated the site, followed by additional waves. The tsunami caused extensive damage to site facilities and resulted in a complete loss of all ac electrical power at Units 1 through 5, a condition known as station blackout. In addition, all direct current electrical power was lost early in the event on Units 1 and 2 and for some period of time at the other units. Unit 6 retained the function of one air-cooled EDG. Despite their actions, the operators lost the ability to cool the fuel in the Unit 1 reactor after several hours, in the Unit 2 reactor after about 70 hours, and in the Unit 3 reactor after about 36 hours, resulting in damage to the nuclear fuel shortly after the loss of cooling capabilities.

The Unit 4 spent fuel pool contained the highest heat load of the six units with the full core present in the spent fuel pool and the refueling gates installed. However, because Unit 4 had been shut down for more than 3 months, the heat load was low relative to that present in spent fuel pools in the United States following shutdown for reactor refueling. Following the earthquake and tsunami, the operators in the Units 3 and 4 control room focused their efforts on stabilizing the Unit 3 reactor. During the event, concern grew that the spent fuel was overheating, causing a high-temperature reaction of steam and zirconium fuel cladding generating hydrogen gas. This concern persisted primarily due to a lack of readily available and reliable information on water levels in the spent fuel pools. Helicopter water drops, water cannons, and cement delivery vehicles with articulating booms were used to refill the pools, which diverted resources and attention from other efforts. Subsequent analysis determined that the water level in the Unit 4 spent fuel pool did not drop below the top of the stored fuel and no significant fuel damage occurred. The lack of information on the condition of the spent fuel pools contributed to a poor understanding of possible radiation releases and adversely impacted effective prioritization of emergency response actions by decision makers.

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

As directed by the Commission's Staff Requirements Memorandum (SRM) for SECY-11-0093, the NRC staff reviewed the NTTF recommendations within the context of the NRC's existing regulatory framework and considered the various regulatory vehicles available to the NRC to implement the recommendations. SECY-11-0124 and SECY-11-0137 established the NRC staff's prioritization of the recommendations based upon the potential safety enhancements.

Current regulatory requirements and existing plant capabilities allow the NRC to conclude that a sequence of events such as the Fukushima Dai-ichi accident is unlikely to occur in the United States. Therefore, continued operation and continued licensing activities do not pose an imminent threat to public health and safety. However, the NRC's assessment of new insights from the events at Fukushima Dai-ichi leads the NRC staff to conclude that additional requirements must be imposed on Licensees and CP holders to increase the capability of nuclear power plants to mitigate beyond-design-basis external events. These additional requirements are needed to provide adequate protection to public health and safety, as set forth in Section III of this Order.

Additional details on an acceptable approach for complying with this Order will be contained in final interim staff guidance (ISG) scheduled to be issued by the NRC in August 2012. This guidance will include a template to be used for the plan that will be submitted in accordance with Section IV, Condition C.1 below.

III.

Reasonable assurance of adequate protection of public health and safety and assurance of the common defense and security are the fundamental NRC regulatory objectives. Compliance with NRC requirements plays a critical role in giving the NRC confidence that Licensees and CP holders are maintaining an adequate level of public health and safety and common defense and security. While compliance with NRC requirements presumptively ensures adequate protection, new information may reveal that additional requirements are warranted. In such situations, the Commission may act in accordance with its statutory authority under Section 161 of the Atomic Energy Act of 1954, as amended, to require Licensees and CP holders to take action in order to protect health and safety and common defense and security.

To protect public health and safety from the inadvertent release of radioactive materials, the NRC's defense-in-depth strategy includes multiple layers of protection: (1) prevention of accidents by virtue of the design, construction, and operation of the plant; (2) mitigation features to prevent radioactive releases should an accident occur; and (3) emergency preparedness programs that include measures such as sheltering and evacuation. The defense-in-depth strategy also provides for multiple physical barriers to contain the radioactive materials in the event of an accident. The barriers are the fuel cladding, the reactor coolant pressure boundary, and the containment. These defense-in-depth features are embodied in the existing regulatory requirements and thereby provide adequate protection of public health and safety.

In the case of spent fuel pools, compliance with existing regulations and guidance presumptively provides reasonable assurance of the safe storage of spent fuel. In particular, Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes the general design criteria (GDC) for nuclear power plants. All currently operating reactors were licensed to the GDC or meet the intent of the GDC. The GDC provide the design features of the spent fuel storage and handling systems and the protection of these systems from natural phenomena and operational events. The accidents considered during licensing of U.S. nuclear power plants typically include failure of the forced cooling system and loss of spent fuel pool inventory at a specified rate within the capacity of the makeup water system. Further, spent fuel pools at U.S. nuclear power plants rely on maintenance of an adequate inventory of water under accident conditions to provide containment, as well as the cooling and shielding safety functions.

During the events in Fukushima, responders were without reliable instrumentation to determine water level in the spent fuel pool. This caused concerns that the pool may have boiled dry, resulting in fuel damage.[2] Fukushima demonstrated the confusion and misapplication of resources that can result from beyond-design-basis external events when adequate instrumentation is not available.

The spent fuel pool level instrumentation at U.S. nuclear power plants is typically narrow range and, therefore, only capable of monitoring normal and slightly off-normal conditions. Although the likelihood of a catastrophic event affecting nuclear power plants and the associated spent fuel pools in the United States remains very low, beyond-design-basis external events could challenge the ability of existing instrumentation to provide emergency responders with reliable information on the condition of spent fuel pools. Reliable and available indication is essential to ensure plant personnel can effectively prioritize emergency actions.

Accordingly, the NRC has concluded that there is a need to redefine the level of protection of public health and safety regarded as adequate under the provisions of the backfit rule, 10 CFR 50.109(a)(4)(iii), and is requiring actions of Licensees and CP holders to meet the new level of protection. In addition, pursuant to 10 CFR 2.202, the NRC finds that the public health, safety and interest require that this Order be made immediately effective.

The Commission has determined that adequate protection of public health and safety requires that all power reactor Licensees and CP holders have a reliable means of remotely monitoring wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event. These new requirements provide a greater capability, consistent with the overall defense-in-depth philosophy, and therefore greater assurance that the challenges posed by beyond-design-basis external events to power reactors do not pose an undue risk to public health and safety. In order

[2] *See Institute of Nuclear Power Operations (INPO) 11-005, "Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station," Revision 0, issued November 2011, p. 36.*

to provide reasonable assurance of adequate protection of public health and safety, all operating reactor licenses and CPs under Part 50 identified in Attachment 1 to this Order shall be modified to include the requirements identified in Attachment 2 to this Order. All combined licenses (COLs) under Part 52 identified in Attachment 1 to this Order shall be modified to include the requirements identified in Attachment 3 to this Order.

IV.

Accordingly, pursuant to Sections 161b, 161i, 161o, and 182 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.202, and 10 CFR Parts 50 and 52, IT IS HEREBY ORDERED, EFFECTIVE IMMEDIATELY, THAT ALL LICENSES AND CONSTRUCTION PERMITS IDENTIFIED IN ATTACHMENT 1 TO THIS ORDER ARE MODIFIED AS FOLLOWS:

A. 1. All holders of CPs issued under Part 50 shall, notwithstanding the provisions of any Commission regulation or CP to the contrary, comply with the requirements described in Attachment 2 to this Order except to the extent that a more stringent requirement is set forth in the CP. These CP holders shall complete full implementation **prior to issuance of an operating license.**

2. All holders of operating licenses issued under Part 50 shall, notwithstanding the provisions of any Commission regulation or license to the contrary, comply with the requirements described in Attachment 2 to this Order except to the extent that a more stringent requirement is set forth in the license. These Licensees shall promptly start implementation of the requirements in Attachment 2 to the Order and shall complete full implementation **no later than two (2) refueling cycles after submittal of the overall integrated plan, as required in Condition C.1.a, or December 31, 2016, whichever comes first.**

3. All holders of COLs issued under Part 52 shall, notwithstanding the provisions of any Commission regulation or license to the contrary, comply with the requirements described in Attachment 3 to this Order except to the extent that a more stringent requirement is set forth in the license. These Licensees shall promptly start implementation of the requirements in Attachment 3 to the Order and shall complete full implementation prior to initial fuel load.

B. 1. All Licensees and CP holders shall, **within twenty (20) days** of the date of this Order, notify the Commission (1) if they are unable to comply with any of the requirements described in Attachment 2 or Attachment 3, (2) if compliance with any of the requirements is unnecessary in their specific circumstances, or (3) if implementation of any of the requirements would cause the Licensee or CP holder to be in violation of the provisions of any Commission regulation or the facility license. The notification shall provide the Licensee's or CP holder's justification for seeking relief from or variation of any specific requirement.

2. Any Licensee or CP holder that considers that implementation of any of the requirements described in Attachment 2 or Attachment 3 to this Order would adversely impact safe and secure operation of the facility must notify the Commission, **within twenty (20) days** of this Order, of the adverse impact, the basis for its determination that the requirement has an adverse impact, and either a proposal for achieving the same objectives specified in the Attachment 2 or Attachment 3 requirement in question, or a schedule for modifying the facility to address the adverse condition. If neither approach is appropriate, the Licensee or CP holder must supplement its response to Condition B.1 of this Order to

identify the condition as a requirement with which it cannot comply, with attendant justifications as required in Condition B.1.

C. 1. a. All holders of operating licenses issued under Part 50 shall **by February 28, 2013**, submit to the Commission for review an overall integrated plan, including a description of how compliance with the requirements described in Attachment 2 will be achieved.

b. All holders of CPs issued under Part 50 or COLs issued under Part 52 shall, **within one (1) year** after issuance of the final ISG, submit to the Commission for review an overall integrated plan, including a description of how compliance with the requirements described in Attachment 2 or Attachment 3 will be achieved.

2. All Licensees and CP holders shall provide an initial status report **sixty (60) days** after the issuance of the final ISG, and **at six (6)-month intervals** following submittal of the overall integrated plan, as required in Condition C.1, which delineates progress made in implementing the requirements of this Order.

3. All Licensees and CP holders shall report to the Commission when full compliance with the requirements described in Attachment 2 or Attachment 3 is achieved.

Licensee or CP holder responses to Conditions B.1, B.2, C.1, C.2, and C.3, above, shall be submitted in accordance with 10 CFR 50.4 and 10 CFR 52.3, as applicable.

As applicable, the Director, Office of Nuclear Reactor Regulation or the Director, Office of New Reactors may, in writing, relax or rescind any of the above conditions upon demonstration by the Licensee or CP holder of good cause.

V.

In accordance with 10 CFR 2.202, the Licensee or CP holder must, and any other person adversely affected by this Order may, submit an answer to this Order, and may request a hearing on this Order, **within twenty (20) days** of the date of this Order. Where good cause is shown, consideration will be given to extending the time to answer or to request a hearing. A request for extension of time in which to submit an answer or request a hearing must be made in writing to the Director, Office of Nuclear Reactor Regulation or to the Director, Office of New Reactors, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. The answer may consent to this Order.

If a hearing is requested by a Licensee, CP holder, or a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearings. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained. Pursuant to 10 CFR 2.202(c)(2)(i), the Licensee, CP holder, or any other person adversely affected by this Order, may, in addition to demanding a hearing, at the time the answer is filed or sooner, move the presiding officer to set aside the immediate effectiveness of the Order on the ground that the Order, including the need for immediate effectiveness, is not based on adequate evidence but on mere suspicion, unfounded allegations, or error.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E-Filing rule (72 FR 49139, August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by telephone at (301) 415-1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/apply-certificates.html*. System requirements for accessing the E-Submittal server are detailed in NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at *http://www.nrc.gov/site-help/esubmittals.html*. Participants may attempt to use other software not listed on the web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E-Filing rule, the participant must file the document using the NRC's online, web-based submission form. In order to serve documents through the Electronic Information Exchange, users will be required to install a web browser plug-in from the NRC web site. Further information on the web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public web site at *http://www.nrc.gov/site-help/esubmittals.html*.

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at *http://www.nrc.gov/site-help/e-submittals.html*. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically using the agency's adjudicatory E-Filing system may seek assistance by contacting the NRC Meta System Help Desk through the "Contact Us" link located on the NRC web site at *http://www.nrc.gov/site-help/e-submittals.html*, by e-mail at MSHD.Resource@nrc.gov, or by a toll-free call at (866) 672-7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using EFiling, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket, which is available to the public at *http://ehd.nrc.gov/EHD_Proceeding/home.asp*, unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

If a person other than the Licensee or CP holder requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.309(d).

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above shall be final twenty (20) days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section IV shall be final when the extension expires if a hearing request has not been received. AN ANSWER OR A REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

FOR THE NUCLEAR REGULATORY COMMISSION

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Michael R. Johnson, Director
Office of New Reactors

Dated this ____ day of March 2012

POWER REACTOR LICENSEES AND LICENSEES
WITH ACTIVE AND/OR DEFERRED CONSTRUCTION PERMITS

Arkansas Nuclear One, Units 1 and 2
Entergy Nuclear Operations, Inc.
London, AR
Docket Nos. 50-313 and 50-368
License Nos. DPR-51 and NPF-6

Beaver Valley Power Station, Units 1 and 2
First Energy Nuclear Operating Co.
Shippingport, PA
Docket Nos. 50-334 and 50-412
License Nos. DPR-66 and NPF-73

Bellefonte Nuclear Power Station, Units 1 and 2
Tennessee Valley Authority
Scottsboro, AL
Docket Nos. 50-438 and 50-439
Construction Permit Nos. CPPR-122 and CPPR-123

Braidwood Station, Units 1 and 2
Exelon Generation Co., LLC
Braceville, IL
Docket Nos. 50-456 and 50-457
License Nos. NPF-72 and NPF-77

Browns Ferry Nuclear Plant, Units 1, 2 and 3
Tennessee Valley Authority
Athens, AL
Docket Nos. 50-259, 50-260, and 50-296
License Nos. DPR-33, DPR-52 and DPR-68

Brunswick Steam Electric Plant, Units 1 and 2
Carolina Power & Light Co.
Southport, NC
Docket Nos. 50-325 and 50-324
License Nos. DPR-71 and DPR-62

Byron Station, Units 1 and 2
Exelon Generation Co., LLC
Byron, IL
Docket Nos. 50-454 and 50-455
License Nos. NPF-37 and NPF-66

Callaway Plant
Union Electric Co.
Fulton, MO
Docket No. 50-483
License No. NPF-30

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Calvert Cliffs Nuclear Power Plant, Inc.
Lusby, MD
Docket Nos. 50-317 and 50-318
License Nos. DPR-53 and DPR-69

Catawba Nuclear Station, Units 1 and 2
Duke Energy Carolinas, LLC
York, SC
Docket Nos. 50-413 and 50-414
License Nos. NPF-35 and NPF-52

Clinton Power Station, Unit 1
Exelon Generation Co., LLC
Clinton, IL
Docket No. 50-461
License No. NPF-62

Columbia Generating Station, Unit 2
Energy Northwest
Richland, WA
Docket No. 50-397
License No. NPF-21

Comanche Peak Steam Electric Station, Units 1 and 2
Luminant Generation Co., LLC
Glen Rose, TX
Docket Nos. 50-445 and 50-446
License Nos. NPF-87 and NPF-89

Cooper Nuclear Station
Nebraska Public Power District
Brownville, NE
Docket No. 50-298
License No. DPR-46

Crystal River Nuclear Generating Plant, Unit 3
Florida Power Corp.
Crystal River, FL
Docket No. 50-302
License No. DPR-72

Davis-Besse Nuclear Power Station, Unit 1
First Energy Nuclear Operating Co.
Oak Harbor, OH
Docket No. 50-346
License No. NPF-3

Diablo Canyon Nuclear Power Plant, Units 1 and 2
Pacific Gas & Electric Co.
Avila Beach, CA
Docket Nos. 50-275 and 50-323
License Nos. DPR-80 and DPR-82

Donald C. Cook Nuclear Power Plant, Units 1 and 2
Indiana Michigan Power Co.
Bridgman, MI
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

Dresden Nuclear Power Station, Units 2 and 3
Exelon Generation Co., LLC
Morris, IL
Docket Nos. 50-237 and 50-249
License Nos. DPR-19 and DPR-25

Duane Arnold Energy Center
FPL Energy Duane Arnold, LLC
Palo, IA
Docket No. 50-331
License No. DPR-49

Edwin I. Hatch Nuclear Plant, Units 1 and 2
Southern Nuclear Operating Co.
Baxley, GA
Docket Nos. 50-321 and Docket No. 50-366
License Nos. DPR-57 and NPF-5

Fermi, Unit 2
The Detroit Edison Co.
Newport, MI
Docket No. 50-341
License No. NPF-43

Fort Calhoun Station, Unit 1
Omaha Public Power District
Fort Calhoun, NE
Docket No. 50-285
License No. DPR-40

Grand Gulf Nuclear Station, Unit 1
Entergy Nuclear Operations, Inc.
Port Gibson, MS
Docket No. 50-416
License No. NPF-29

H. B. Robinson Steam Electric Plant, Unit 2
Carolina Power & Light Co.
Hartsville, SC
Docket No. 50-261
License No. DPR-23

Hope Creek Generating Station, Unit 1
PSEG Nuclear, LLC
Hancocks Bridge, NJ
Docket No. 50-354
License No. NPF-57

Indian Point Nuclear Generating Station, Units 2 and 3
Entergy Nuclear Operations, Inc.
Buchanan, NY
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

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

Joseph M. Farley Nuclear Plant, Units 1 and 2
Southern Nuclear Operating Co.
Columbia, AL
Docket Nos. 50-348 and 50-364
License Nos. NPF-2 and NPF-8

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

LaSalle County Station, Units 1 and 2
Exelon Generation Co., LLC
Marseilles, IL
Docket Nos. 50-373 and 50-374
License Nos. NPF-11 and NPF-18

Limerick Generating Station, Units 1 and 2
Exelon Generation Co., LLC
Limerick, PA
Docket Nos. 50-352 and 50-353
License Nos. NPF-39 and NPF-85

McGuire Nuclear Station, Units 1 and 2
Duke Energy Carolinas, LLC
Huntersville, NC
Docket Nos. 50-369 and 50-370
License Nos. NPF-9 and NPF-17

Millstone Power Station, Units 2 and 3
Dominion Nuclear Connecticut, Inc.
Waterford, CT
Docket Nos. 50-336 and 50-423
License Nos. DPR-65 and NPF-49

Monticello Nuclear Generating Plant, Unit 1
Northern States Power Company
Monticello, MN
Docket No. 50-263
License No. DPR-22

Nine Mile Point Nuclear Station, Units 1 and 2
Nine Mile Point Nuclear Station, LLC
Scriba, NY
Docket Nos. 50-220 and 50-410
License Nos. DPR-63 and NPF-69

North Anna Power Station, Units 1 and 2
Virginia Electric & Power Co.
Louisa, VA
Docket Nos. 50-338 and 50-339
License Nos. NPF-4 and NPF-7

Oconee Nuclear Station, Units 1, 2, and 3
Duke Energy Carolinas, LLC
Seneca, SC
Docket Nos. 50-269, 50-270, and 50-287
License Nos. DPR-38, DPR-47, and DPR-55

Oyster Creek Nuclear Generating Station, Unit 1
Exelon Generation Co., LLC
Forked River, NJ
Docket No. 50-219
License No. DPR-16

Palisades Nuclear Plant
Entergy Nuclear Operations, Inc.
Covert, MI
Docket No. 50-255
License No. DPR-20

Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Arizona Public Service Company
Wintersburg, AZ
Docket Nos. 50-528, 50-529, and 50-530
License Nos. NPF-41, NPF-51 and NPF-74

Peach Bottom Atomic Power Station, Units 2 and 3
Exelon Generation Co., LLC
Delta, PA
Docket Nos. 50-277 and 50-278
License Nos. DPR-44 and DPR-56

Perry Nuclear Power Plant, Unit 1
First Energy Nuclear Operating Co.
Perry, OH
Docket No. 50-440
License No. NPF-58

Pilgrim Nuclear Power Station
Entergy Nuclear Operations, Inc.
Plymouth, MA
Docket No. 50-293
License No. DPR-35

Point Beach Nuclear Plant, Units 1 and 2
FPL Energy Duane Arnold, LLC
Two Rivers, WI
Docket Nos. 50-266 and 50-301
License Nos. DPR-24 and DPR-27

Prairie Island Nuclear Generating Plant, Units 1 and 2
Northern States Power Co. Minnesota
Welch, MN
Docket Nos. 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Quad Cities Nuclear Power Station, Units 1 and 2
Exelon Generation Co., LLC
Morris, IL
Docket Nos. 50-254 and 50-265
License Nos. DPR-29 and DPR-30

River Bend Station, Unit 1
Entergy Nuclear Operations, Inc.
St. Francisville, LA
Docket No. 50-458
License No. NPF-47

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

St. Lucie Plant, Units 1 and 2
Florida Power & Light Co.
Jensen Beach, FL
Docket Nos. 50-335 and 50-389
License Nos. DPR-67 and NPF-16

Salem Nuclear Generating Station, Units 1 and 2
PSEG Nuclear, LLC
Hancocks Bridge, NJ
Docket Nos. 50-272 and 50-311
License Nos. DPR-70 and DPR-75

San Onofre Nuclear Generating Station, Units 2 and 3
Southern California Edison Co.
San Clemente, CA
Docket Nos. 50-361 and 50-362
License Nos. NPF-10 and NPF-15

Seabrook Station, Unit 1
FPL Energy Seabrook, LLC
Seabrook, NH
Docket No. 50-443
License No. NPF-86

Sequoyah Nuclear Plant, Units 1 and 2
Tennessee Valley Authority
Soddy-Daisy, TN
Docket Nos. 50-327 and 50-328
License Nos. DPR-77 and DPR-79

Shearon Harris Nuclear Power Plant, Unit 1
Carolina Power & Light Co.
New Hill, NC
Docket No. 50-400
License No. NPF-63

South Texas Project, Units 1 and 2
STP Nuclear Operating Co.
Bay City, TX
Docket Nos. 50-498 and 50-499
License Nos. NPF-76 and NPF-80

Surry Nuclear Power Station, Units 1 and 2
Virginia Electric & Power Co.
Surry, VA
Docket Nos. 50-280 and 50-281
License Nos. DPR-32 and DPR-37

Susquehanna Steam Electric Station, Units 1 and 2
PPL Susquehanna, LLC
Salem Township, Luzerne Co., PA
Docket Nos. 50-387 and 50-388
License Nos. NPF-22 and NPF-14

Three Mile Island Nuclear Station, Unit 1
Exelon Generation Co., LLC
Middletown, PA
Docket No. 50-289
License No. DPR-50

Turkey Point Nuclear Generating, Units 3 and 4
Florida Power & Light Co.
Homestead, FL
Docket Nos. 50-250 and 50-251
License Nos. DPR-31 and DPR-41

Vermont Yankee Nuclear Power Plant, Unit 1
Entergy Nuclear Operations, Inc.
Vernon, VT
Docket No. 50-271
License No. DPR-28

Virgil C. Summer Nuclear Station, Unit1
South Carolina Electric & Gas Co.
Jenkinsville, SC
Docket No. 50-395
License No. NPF-12

Vogtle Electric Generating Plant, Units 1, 2, 3, and 4
Southern Nuclear Operating Co.
Waynesboro, GA
Docket Nos. 50-424, 50-425, 52-025, and 52-026
License Nos. NPF-68, NPF-81, NPF-91 and NPF-92

Waterford Steam Electric Station, Unit 3
Entergy Nuclear Operations, Inc.
Killona, LA
Docket No. 50-382
License No. NPF-38

Watts Bar Nuclear Plant, Units 1 and 2
Tennessee Valley Authority
Spring City, TN
Docket No. 50-390 and 50-391
License No. NPF-90 and
Construction Permit No. CPPR-92

Wolf Creek Generating Station, Unit 1
Wolf Creek Nuclear Operating Corp.
Burlington, Coffey County, KS
Docket No. 50-482
License No. NPF-42

REQUIREMENTS FOR RELIABLE SPENT FUEL POOL LEVEL INSTRUMENTATION AT OPERATING REACTOR SITES AND CONSTRUCTION PERMIT HOLDERS

All licensees identified in Attachment 1 to this Order shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

1. The spent fuel pool level instrumentation shall include the following design features:

    1.1. Instruments: The instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel. The backup instrument channel may be fixed or portable. Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor spent fuel pool water level under conditions that restrict direct personnel access to the pool, such as partial structural damage, high radiation levels, or heat and humidity from a boiling pool.

    1.2 Arrangement: The spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the primary instrument channel and fixed portions of the backup instrument channel, if applicable, to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.

    1.3 Mounting: Installed instrument channel equipment within the spent fuel pool shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.

    1. 4 Qualification: The primary and backup instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. This reliability shall be established through use of an augmented quality assurance process (e.g., a process similar to that applied to the site fire protection program).

    1.5 Independence: The primary instrument channel shall be independent of the backup instrument channel.

    1. 6 Power supplies: Permanently installed instrumentation channels shall each be powered by a separate power supply. Permanently installed and portable instrumentation channels shall provide for power connections from sources independent of the plant ac and dc power distribution systems, such as portable generators or replaceable batteries. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.

1. 7 Accuracy: The instrument channels shall maintain their designed accuracy following a power interruption or change in power source without recalibration.

1. 8 Testing: The instrument channel design shall provide for routine testing and calibration.

1. 9 Display: Trained personnel shall be able to monitor the spent fuel pool water level from the control room, alternate shutdown panel, or other appropriate and accessible location. The display shall provide on-demand or continuous indication of spent fuel pool water level.

2. The spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of the following programs:

2.1 Training: Personnel shall be trained in the use and the provision of alternate power to the primary and backup instrument channels.

2.2 Procedures: Procedures shall be established and maintained for the testing, calibration, and use of the primary and backup spent fuel pool instrument channels.

2.3 Testing and Calibration: Processes shall be established and maintained for scheduling and implementing necessary testing and calibration of the primary and backup spent fuel pool level instrument channels to maintain the instrument channels at the design accuracy.

REQUIREMENTS FOR RELIABLE SPENT FUEL POOL LEVEL
INSTRUMENTATION AT COL HOLDER REACTOR SITES

Attachment 2 to this Order for Part 50 Licensees requires reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

The design bases of Vogtle Units 3&4 address many of these attributes of spent fuel pool level instrumentation. The NRC staff reviewed these design features prior to issuance of the combined licenses for these facilities and certification of the AP1000 design referenced therein. The AP1000 certified design largely addresses the requirements in Attachment 2 by providing two safety-related spent fuel pool level instrument channels. The instruments measure level from the top of the spent fuel pool to the top of the fuel racks to address the range requirements listed above. The safety-related classification provides for the following additional design features:

- Seismic and environmental qualification of the instruments
- Independent power supplies
- Electrical isolation and physical separation between instrument channels
- Display in the control room as part of the post-accident monitoring instrumentation
- Routine calibration and testing

As such, this Order requires Vogtle Units 3&4 to address the following requirements that were not specified in the certified design.

1. The spent fuel pool level instrumentation shall include the following design features:

    1.1 Arrangement: The spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the safety-related instruments to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.

    1.2 Qualification: The level instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period.

    1.3 Power supplies: Instrumentation channels shall provide for power connections from sources independent of the plant alternating current (ac) and direct current (dc) power distribution systems, such as portable generators or replaceable batteries. Power supply designs should provide for quick and accessible connection of sources independent of the plant ac and dc power distribution systems. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.

1.4 Accuracy: The instrument shall maintain its designed accuracy following a power interruption or change in power source without recalibration.

1.5 Display: The display shall provide on-demand or continuous indication of spent fuel pool water level.

2. The spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of a training program. Personnel shall be trained in the use and the provision of alternate power to the safety-related level instrument channels.

[Addressee]

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

Dear [Name]:

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's 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 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 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 Access 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 emergency preparedness.

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 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.

[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 is covered by the Office of Management and Budget (OMB) clearance number 3150-0011, which expires October 21, 2014. The estimated reporting burden for this collection of information is detailed in Table 1. This estimate includes the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, 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 on any aspect of this information collection, including suggestions for reducing the burden, to the Records and FOIA/Privacy Services Branch (T5-F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet electronic mail to infocollects@nrc.gov; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503. The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it 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 their 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 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 50.54(f) letters to all licensees to request 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 5 years. It is anticipated that collection of this information for all plants will take no longer than 7 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 1 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 reinforced the need for effective EP, the objective of which is to ensure the capability to implement adequate 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 SBO 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

Docket Nos.

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: EMERGENCY PREPAREDNESS]

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

Sincerely,

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5. [RECOMMENDATION 9.3: EMERGENCY PREPAREDNESS]

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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

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

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 (IN) 2010-018, “Generic Issue 199, ‘Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,’” dated September 2, 2010 (ADAMS Accession No. ML101970221), summarizes the results of the GI-199 safety/risk assessment.

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 have been revised to 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 present-day NRC process for determining the seismic design basis ground motions, 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 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.

[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.

- 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.

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 be at least be a Level 1 SPRA 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, 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 spent fuel pool. The evaluation should consider SFP walls, liner, penetrations (cooling water supplies or returns, drains), transfer gates and seals, seals and bellows between the spent fuel pool, 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):

Seismic Hazard Evaluation

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

(2) site-specific, performance-based ground motion response spectrum (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)

Seismic Risk Evaluation

(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) Spent Fuel Pool 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 a risk assessment approach, including acceptance criteria[2].

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 assessment 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 acceptability of the licensee's proposed last addressee submittal, the NRC staff will have determined the risk assessment 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 assessment (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 assessment (items 8A or 8B and 9 above) over a period not to exceed 4 years from the date of the prioritization.

[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.

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.

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 (1×10^{-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 (1×10^{-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.

[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.

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

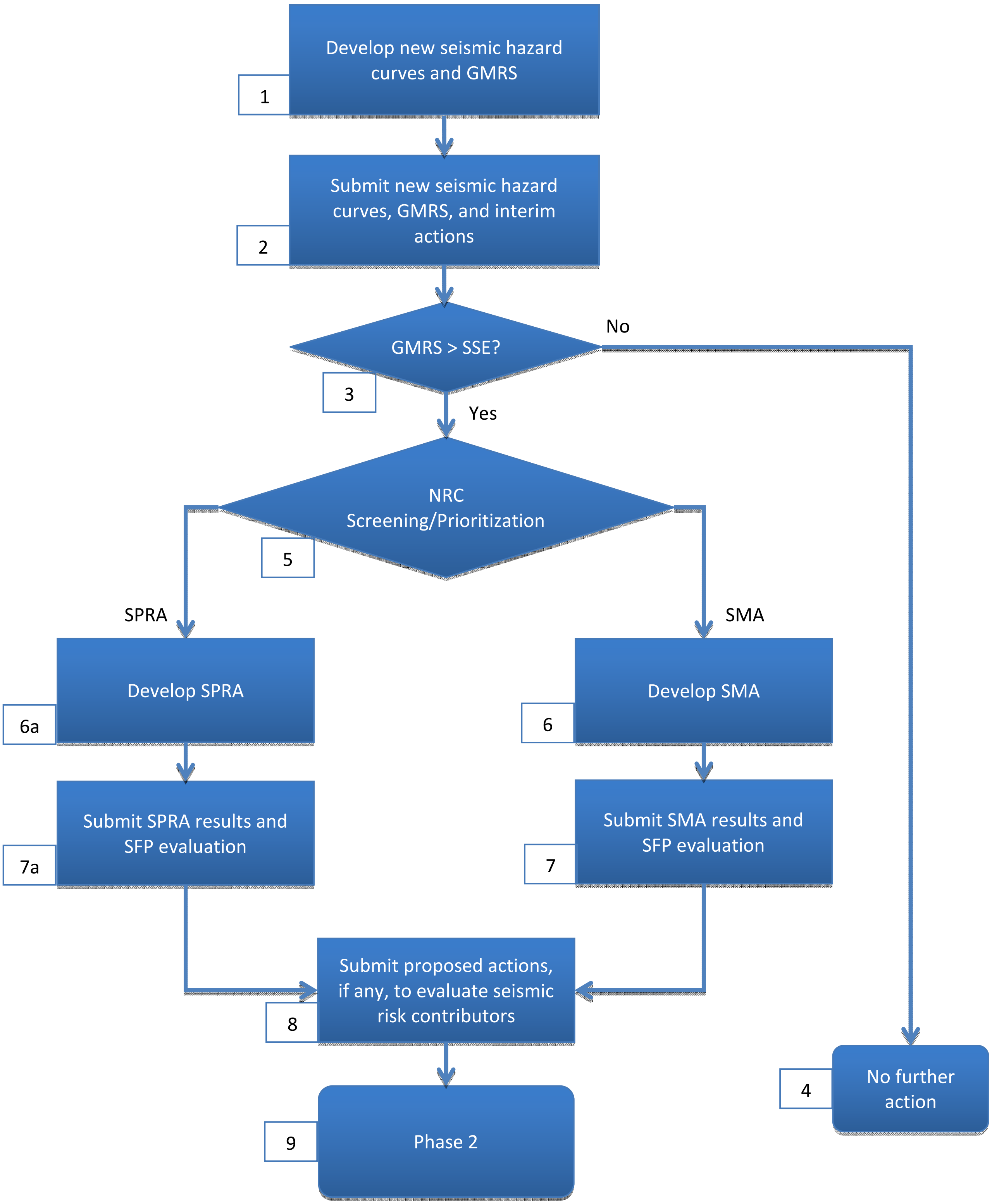

Figure 1. Development of Requested Information and Its Use in Regulatory Analysis.

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,” ML11272A111, October 3, 2011.

SECY 11-0124, “Recommended Action to be taken without Delay from the Near-Term Task Force Report,” ML11245A158, September 9, 2011.

10 CFR 50.54(f) – “Conditions of Licenses”

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“Generic Issue (GI)-199, ‘Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,’” 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),” ML031150485, June 28, 1991.

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

“Identification of a Generic Seismic Issue,” 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.’” 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,’” 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,’” 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,’” 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,’” 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,” ML042820192, September 30, 2004.

“Draft NRC Generic Letter 2011-XX: Seismic Risk Evaluations for Operating Reactors,” ML111710783, July 26, 2011.

NUREG/CR-4334, “An Approach to the Quantification of Seismic Margins in Nuclear Power Plants,” 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, ML090410014, March 2009.

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

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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.

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NUREG/CR-6728, “Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines,” ML013100232, October 2001.

Regulatory Guide 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” 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 proposed Generic Issue (GI) on 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

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 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 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 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 quantative approach to assess the flooding hazard. NUREG-1742, “Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program,” 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 Generic Letter (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."

New Requirements for Evaluation of Dam Hazards in 10 CFR 100.20:

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.

Tsunami and Regulatory Guide 1.59 Updates:

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 are 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 operating license (COL) applications.

Proposed Generic Issue on Upstream Dam Failures:

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 staff evaluation of this is the proposed GI ongoing. The NRC staff anticipates that the information gathered by this request will likely be applicable to evaluation of the GI as well.

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

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.

Present-day regulations for reactor site criteria (Subpart B to 10 CFR Part 100) state, 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 (Section 100.21(d)). Factors to be considered when evaluating sites includes the nature and proximity of dams and other man-related hazards (Section 100.20(b)) and the physical characteristics of the site, including the hydrology (Section 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’s 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 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 time frame 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 that could be adversely affected by the flood conditions and lead to degradation of the flood protection (the loss of ultimate heat sink 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 ultimate heat sink 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

[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’s and industry’s 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

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[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 walk downs

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 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.

(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.

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

(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

NRC regulations require that structure, systems and components 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 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 US 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, 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, 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.

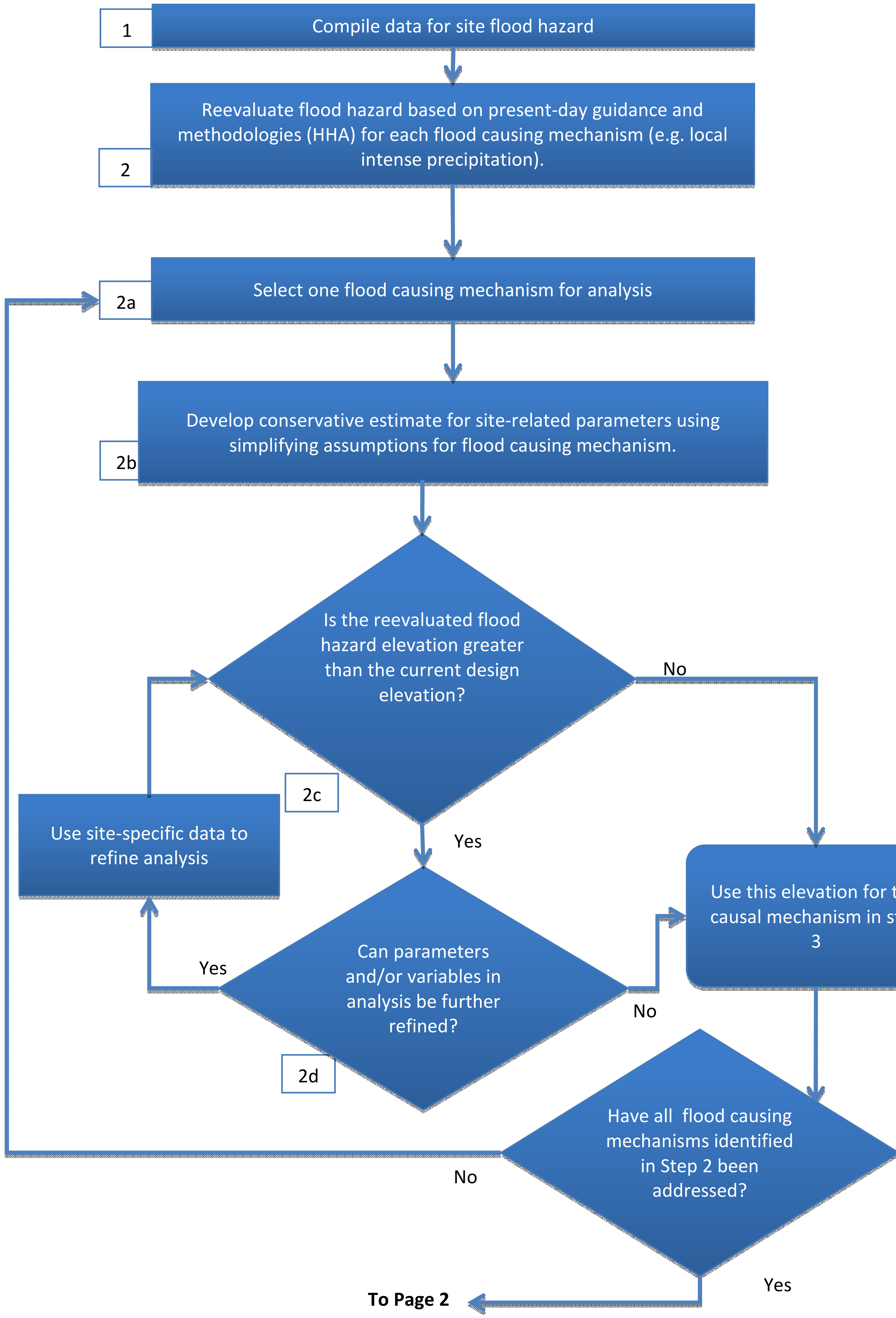

Figure 1. Development of Requested Information and Its Use in Regulatory Analysis. Page 1 of 2

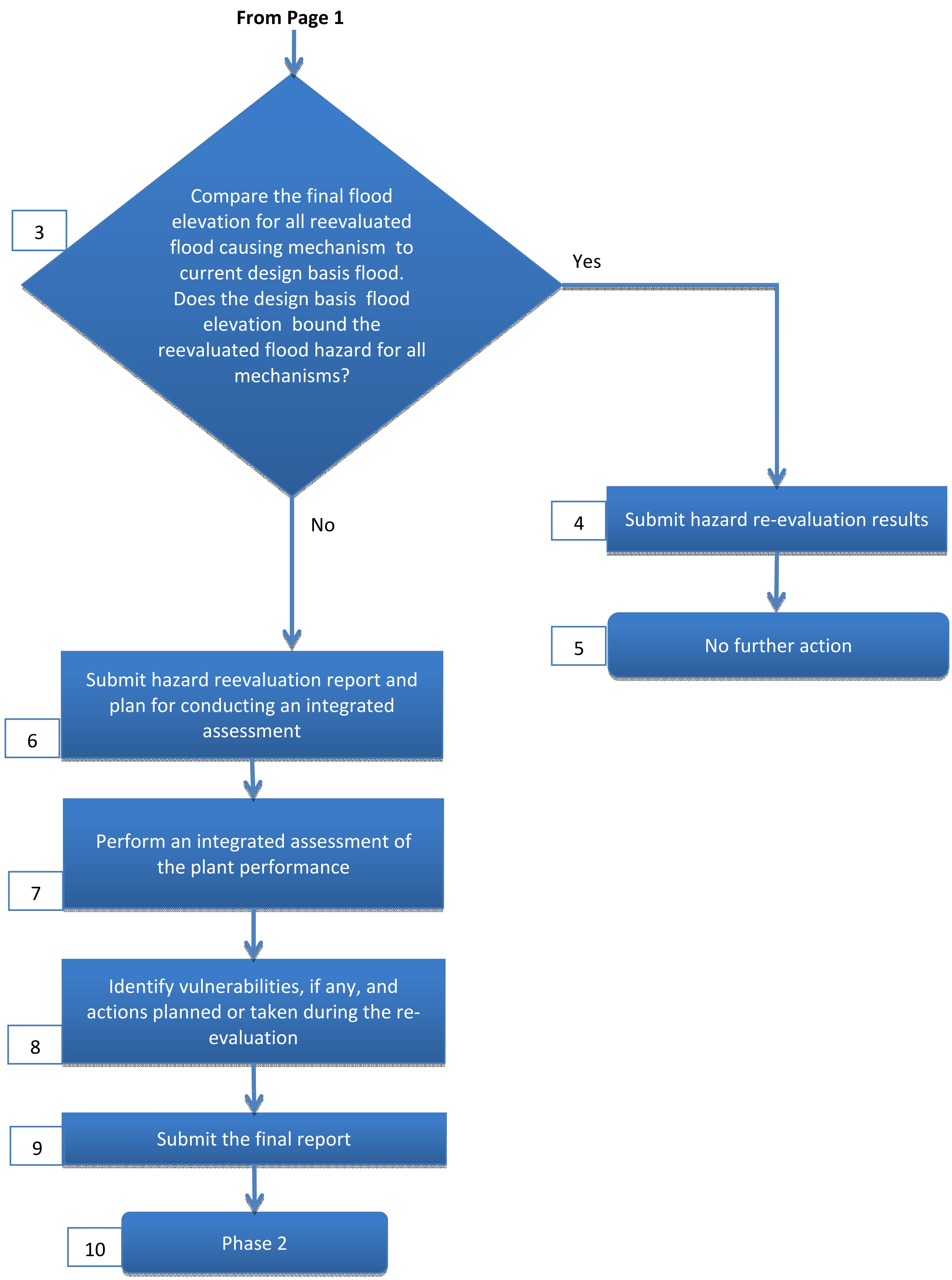

Figure 1. Development of Requested Information and Its Use in Regulatory Analysis. Page 2 of 2

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," ML11245A158, September 9, 2011.

SECY 11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," 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," 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," 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," ML042820192, September 30, 2004.

SRM SECY 11-0124, "Recommended Actions To Be Taken Without Delay from the Near-Term Task Force Report," ML112911571, October 18, 2011.
SRM SECY 11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," 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)," 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." ML11321A195, November 2011.

RG 1.29, "Seismic Design Classification," Revision 4, ML070310052, March 2007.

RG 1.59, "Design Basis Floods for Nuclear Power Plants," Revision 2, ML003740388, August 1977.

RG 1.102, "Flood Protection for Nuclear Power Plants," Revision 1, 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, non-conforming, 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 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 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 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, non-conforming, 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, "Followup 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 Near-Term Task Force recommendations as described in the SECY. With regard to Recommendation 2.3, the 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 seismic walkdowns to be endorsed by the NRC staff following interactions with external stakeholders, perform seismic walkdowns to identify and address plant-specific degraded, non-conforming, 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.

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 (SBO) 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, non-conforming, 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, non-conforming, 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, non-conforming, 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, non-conforming, or unanalyzed condition (i.e., non-compliance 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, Rev 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

[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.

(g.) Procedures used to evaluate the effectiveness of the monitoring and maintenance programs
(h.) Procedures used to evaluate the passive protection systems
(i.) Procedures used to evaluate active protection systems (operator availability, operator training, timeliness of response, equipment maintenance and operability, back-up availability, operator access under various site conditions)
(j.) Procedures and acceptance criteria used for determining the viability of protection measures including mitigation strategies
(k.) Maintenance and reliability of mitigation or protection systems including the UHS
(l.) 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, non-conforming, 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, Rev 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 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," ML11245A158, September 9, 2011.

SECY 11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," ML11272A111, October 3, 2011.

SRM SECY 11-0124, "Recommended Action to be taken without Delay from the Near-Term Task Force Report," ML112911571, dated October 18, 2011.

SRM SECY 11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," 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, non-conforming, 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 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, non-conforming, 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 identifing 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, non-conforming, 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, non-conforming, 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.

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 conincident physical damage to non-safety 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, non-conforming, 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.

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

[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.

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, non-conforming, or unanalyzed condition (i.e., non-compliance 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, Rev 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 for Example 3.1.3 of 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.

(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, non-conforming, 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, Rev 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.

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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.

(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 Accession No. ML11245A158, dated September 9, 2011.

SECY-11-0137, “Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons-Learned,” 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,” 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, 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, 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 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 SBO.

APPLICABLE REGULATORY REQUIREMENTS AND GUIDANCE

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

1. 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, Tokoku 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 multi-unit event, with a loss of all AC power.

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.

APPLICABLE REGULATORY REQUIREMENTS AND GUIDANCE

- 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:

> 5. 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 (EOF) that is capable of handling multi-unit events, the onsite technical support center (TSC) and operational support center (OSC) 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 (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 (ML1113010523). 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 (ERO) 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 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 units

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 NTTF Recommendation 9.3.

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 interim staff guidance 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.

Six-Month Status Update on Other Charter Activities

This is the U.S. Nuclear Regulatory Commission (NRC) staff's first 6-month periodic update on the review work conducted under the Charter in accordance with Staff Requirements Memorandum (SRM)-SECY-11-0117, "Charter for the Nuclear Regulatory Commission Steering Committee to Conduct a Longer-Term Review of the Events in Japan." This includes highlights of any potential policy issues that have arisen for Commission consideration and recommendations regarding the sunset of the Steering Committee, the Advisory Committee, and the Project Directorate.

Accident Timeline

The staff continues to receive specific information on the sequence of events and the status of equipment throughout the accident at Fukushima Daiichi. Specific documented sources include the following:

- Nuclear Emergency Response Headquarters—Government of Japan, "Report of the Japanese Government to the IAEA Ministerial Conference on Nuclear Safety—The Accident at TEPCO's Fukushima Nuclear Power Stations," International Atomic Energy Agency (IAEA) Ministerial Conference on Nuclear Safety, Vienna, Austria, June 7, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11178A379)

- Institute of Nuclear Power Operations (INPO) 11-05, "Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station," Revision 0, issued November 2011 (ADAMS Accession No. ML11347A454)

- Executive Summary of the Interim Report, Investigation Committee on the Accidents at Fukushima Nuclear Power Stations of Tokyo Electric Power Company, December 26, 2011 (http://icanps.go.jp/eng/111226ExecutiveSummary.pdf)

These reports validate the staff's basic understanding of events as presented in the Near-Term Task Force (NTTF) report, dated July 12, 2011, report, and continue to support the staff's plan for regulatory action. The staff will continue to follow the development of a more detailed timeline of events to support these and longer-term actions.

As noted in SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, the NRC and the U.S. Department of Energy signed a Fukushima Daiichi Accident Study addendum to its memorandum of understanding on Cooperative Nuclear Safety Research (ADAMS Accession No. ML111930010) in July 2011. This cooperative research program will, among other things, develop a detailed understanding of the accident progression of each reactor and spent fuel pool. The staff also continues to work with Federal counterparts, industry, and the international community, including the Government of Japan, to establish cooperative efforts to share and integrate specific information into a common understanding of the sequence of events at the Fukushima Daiichi facility.

Ongoing Tier 1, 2, and 3 Regulatory Actions, Additional Issues, and Advisory Committee on Reactor Safeguards Recommendations

The staff continues work on Tier 1 and 2 regulatory actions in a manner that is consistent with the milestone schedule set forth in SECY-11-0137 and SRM-SECY-11-0124, "Staff Requirements-SECY-11-0124-Recommended Actions to be Taken without Delay from the Near-Term Task Force Report," as modified by this paper.

As described in Enclosure 3 of this paper, the staff developed a process for addressing additional issues that arise as a result of ongoing interactions with both domestic and international stakeholders, advisory committee recommendations, and internal staff deliberations. This process includes vetting documented issues by a screening group of agency senior-level scientists and engineers. This group makes recommendations to the Steering Committee on whether each issue is valid and has a nexus to the Fukushima Dai-ichi accident, or should be dispositioned with no additional action or some other NRC process, such as the generic issues resolution process.

NTTF Recommendation 4.1
Station Blackout (SBO) regulatory actions (Tier 1)

The staff has developed an Advanced Notice of Proposed Rulemaking (ANPR) soliciting external stakeholder input regarding regulatory activities for SBO mitigation. The ANPR is currently in concurrence with the review and approval effort occurring in parallel with this SECY paper. It is expected that the EDO will sign and issue this ANPR in the near term. The staff plans to hold a category 3 public meeting during the ANPR comment period. The meeting is not intended for the NRC to receive comments and instead is for the NRC to discuss the ANPR with external stakeholders to inform stakeholder views on SBO mitigation and thereby support stakeholders providing written feedback in response to the ANPR.

NTTF Recommendation 7.2, 7.3, 7.4, 7.5
Rulemaking to provide reliable spent fuel pool instrumentation and makeup capabilities (Tier 2)

This rulemaking will follow the staff's issuance of the proposed order that requires reliable instrumentation in spent fuel pools. The staff is budgeting resources and assessing the availability of staff with the necessary skills to develop a technical basis for a rulemaking that may begin in late calendar year 2012.

NTTF Recommendation 8
Integration of Onsite Emergency Response Processes, Procedures, Training and Exercises (Tier 1)

The development of the NTTF Recommendation 8 ANPR is underway. The working group is planning to hold public meetings to obtain stakeholder input on the proposed rulemaking strategies.

Comparison of Japanese and U.S. Requirements for Station Blackout

Upon review of Japan's ministerial orders and guides, the staff concludes that Japanese regulations require nuclear power plants to be designed such that safe shutdown of the reactor can be ensured in case of a short-term station blackout. The staff also finds that the regulatory expectations for station blackout mitigation are similar between the two countries.

Recommendation 1

In an SRM on SECY-11-0093, “Near-Term Report and Recommendations for Agency Actions Following the Events in Japan,” dated August 19, 2011, the Commission directed that NTTF Recommendation 1 should be pursued independent of any activities associated with the review of the other Task Force recommendations. To implement this direction, the staff established a working group to develop a comprehensive set of options for the Commission, including resource estimates, and the staff’s recommendation. This activity is currently scheduled to be completed in February 2013 and will be coordinated with a number of ongoing staff activities related to defense in depth and regulatory framework, including the following:

- the Chairman’s Risk-Informed Regulations Task Force

- updates to Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” to address defense in depth

- technology-neutral framework approach—NUREG-1860, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing”

Improving Communication with Stakeholders

The staff’s efforts to improve communication with stakeholders are summarized in SECY-12-0010, “Engagement of Stakeholders Regarding the Events in Japan,” which includes a description of the staff’s progress and further recommendations on developing a chronology of events suitable for the general public, and to consulting with individual public citizens on the readability of the NTTF report.

Policy Issues

Additional policy issues identified by the staff are addressed in the body of this paper. This includes the staff’s plans to submit to the Commission in July 2012 a notation vote paper that addresses operability of containment vents under severe accident conditions, the addition of filters to containment vents, and the addition of vents in areas outside the reactor building.

Plans to Sunset Longer Term Review Organization

In SRM-SECY-11-0117, the Commission specified that the longer term review will conclude when all longer term evaluations have been completed and regulatory actions identified and those regulatory actions have been referred to the NRC line organization for action using existing processes (e.g., the rulemaking process). Within the rubric of SECY-11-0137, the staff anticipates that completion of longer term evaluations will be marked by the completion of the staff’s evaluation of the schedule and milestones, resources and critical skill sets, and implementation challenges related to addressing the Tier 3 recommendations. A Commission paper on Tier 3 recommendations is due to the Commission in early July 2012. The staff will provide more detailed plans for sunsetting the longer term review organization in its paper on Tier 3 recommendations.

National Academy of Sciences Study

The Conference Report on the Consolidated Appropriations Act, 2012 (Public Law 112-74) directs the NRC to transfer $2,000,000 to the National Academy of Sciences (NAS) to fund an NAS study of the lessons learned from the events at the Fukushima nuclear plant. The project plan and budget for this study have been finalized and the funds have been transferred to NAS. The staff is working closely with NAS in anticipation of the study starting in the near term.

Full-scale Seismic and Kinetic Impact Tests

The Senate Report[1] on a draft version of Public Law 112-74 includes the following direction to NRC:

> The Committee is concerned that risks to public health and safety exist due to a lack of understanding how critical nuclear energy infrastructure, particularly storage ponds and containers for spent nuclear fuel and waste, will respond to a catastrophic earthquake or kinetic impact event. The Committee directs the Nuclear Regulatory Commission [NRC] to develop protocols for the use of existing domestic seismic testing facilities, including the National Science Foundation's National Earthquake Engineering Simulation [NEES] program, to conduct tests on full-scale specimens of critical nuclear infrastructure, in order to validate related computer models and inform subsequent mitigation strategies. The NRC shall collaborate with NEES to submit a related plan and proposed budget to the Committee by January 23, 2012.

The Senate Report was completed on September 7, 2011, over 3 months before the President signed Public Law 112-74 on December 23, 2011. Therefore, the staff is in discussions with Senate staff regarding a revised schedule for the plan and proposed budget related to this action.

Resource Estimate and Schedule for Probabilistic Risk Analysis Methodology on Seismically Induced Fires and Floods

Background

As described in the NTTF Report, seismically induced fires have the potential to cause multiple failures of safety-related systems and induce separate fires in multiple locations at the site. Additionally, it has been recognized that events such as pipe ruptures (and subsequent flooding) could cause such problems in multiple locations simultaneously. Although these issues have been examined to a limited degree in the Generic Issues Program and Generic Letter (GL) 88-20, Supplement 5, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” the NTTF concluded that the staff should reevaluate the potential for common-mode failures of plant safety equipment as the result of seismically induced fires and floods. Although this recommendation (NTTF Recommendation 3) was categorized as a Tier 3 item (identified for long-term evaluation), SRM-SECY-11-0137 directed the staff to initiate a probabilistic risk assessment (PRA) methodology to evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods as part of Tier 1 activities. Furthermore, the staff was asked to include a discussion of the resource

[1] S. Rep. No. 112-75 (Sep. 7, 2011)

estimate and schedule to develop the PRA methodology in the next 6-month status update to the Commission, as required by SRM-SECY-11-0117.

Staff Recommendation

The staff recognizes that the development of a PRA methodology to address seismically induced fires and floods represents a complex challenge. The scope of this effort is expected to cover seismically induced fires internal to the nuclear power plant, internal seismically induced floods (e.g., piping and tank ruptures), external seismically induced floods (e.g., upstream dam failures), and seismically induced losses of heat sink (e.g., downstream dam failures). There are significant challenges associated with this effort including, but not limited to the following:

- hazard definition and characterization
    - quantification of seismically induced fire ignition
    - quantification of site-specific seismically induced flooding frequencies
    - treatment of uncertainties
- modeling concurrent and subsequent initiating events
- treatment of systems interactions
- human reliability analysis applicability to seismically induced hazards
- multiunit risk considerations

The staff intends to engage in a variety of preplanning activities over the next four months in order to lay a foundation for the development of a more detailed and complete plan to address seismically induced fires and floods. Specific preplanning activities include the following:

1. Define specific objectives of the methodology:

    a. the purpose of the method (e.g., screening and/or detailed analysis)

    b. the anticipated scope of the method (e.g., operational modes, inclusion/exclusion of spent fuel pools)

    c. potential risk criteria to be used in terms of assessing enhancements to the capability to prevent or mitigate seismically induced fires and floods

    d. intended users (NRC staff and/or industry)

2. Identify internal and external stakeholders and assess their level of needed involvement for the development of the PRA methodology.

3. To the extent practical, gather relevant information, including nuclear power plant operating experience, general seismic experience, international data, and academic research.

4. As practical, coordinate planning activities with other initiatives, such as:

    a. post-Fukushima request for information letters (under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(f))

   b. other related research activities, including generic issue resolution and standardized plant analysis risk

   c. external hazard model development

5. Estimate resources required to develop the detailed project plan (contract and full-time equivalent (FTE) staff).

6. Formulate a schedule for developing the project plan.

The result of this effort will be documented in an initial preplan that will provide a framework for the development of a more detailed project plan to address seismically induced fires and floods.

Challenges

The NRC staff is currently working on a number of issues that would need to be integrated into the development of this PRA methodology. For example, the staff is addressing several generic issues related to this topic, including the following:

- GI-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants"

Additionally, the issuance of 10 CFR 50.54(f) letters presented in this paper and subsequent licensee responses should be considered in the development of the PRA methodology. In particular, the response to NTTF Recommendations 2.1 and 2.3 have the potential to provide additional insights into seismic and flooding hazard characterization which, in turn, may affect both the methodology and the input information to correctly assess potential enhancements to the capability to mitigate such events. It is also recognized that the manner in which licensees respond to these 10 CFR 50.54(f) letters may have implications for the implementation of the PRA methodology (e.g., use of seismic margins analysis or seismic PRA).

There are very few members of the staff with the requisite knowledge, skills, and abilities in seismic, fire, and flooding PRA to efficiently perform the above pre-planning activities. These staff members are currently engaged in other high priority work supporting post-Fukushima activities and development of agency PRA models for external hazards and fire. Consequently, the amount of staff resources that can be applied to the pre-planning effort for the development of a PRA method for seismically induced fires and floods are limited. This will reduce the level of detail and technical depth that the staff can include in the initial pre-plan.

Resources

The staff anticipates that it would have approximately 0.1 FTE available over the next four months to develop an initial pre-plan to support the later formulation of a detailed project plan for the development of a PRA methodology to address seismically induced fires and floods. No contract resources are anticipated for this preplanning effort.

Deliverables

1. Initial pre-plan document that will provide a framework for the development of a more detailed project plan

Schedule

1.	Complete the initial pre-plan:	June 2012
2.	Provide status in next SECY paper update	July 2012