**DRAFT - WHITE PAPER**
**PROPOSED PLANS FOR RESOLVING OPEN FUKUSHIMA**
**TIER 2 AND 3 RECOMMENDATIONS**

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## Summary

This draft white paper is provided to support discussion during meetings between the NRC and industry Fukushima steering committees and the Advisory Committee on Reactor Safeguards (ACRS) on the proposed plans to resolve the Fukushima Tier 2 and 3 recommendations. These are the lower-priority portion of the recommendations developed in response to the March 11, 2011, accident at the Fukushima Dai-ichi nuclear power plant. The NRC staff is proposing these plans to support more timely consideration of Fukushima lessons learned, to support agency planning, and to provide regulatory stability to the industry. The staff's approach to resolve the open Tier 2 and 3 recommendations primarily involved technical evaluations to assess the need for additional regulatory action, considering the existing protection provided under current NRC requirements and new information obtained during completion of Tier 1 (high priority) activities. Proposed actions include closing those recommendations that have achieved sufficient progress, interacting with stakeholders on recommendations where beneficial, and providing additional assessment or documentation.

Notwithstanding final resolution of these recommendations, the staff notes that work will continue in some of these areas for many years. For example, the NRC staff will continue to engage with the international community on long-term health studies in the areas around Fukushima Dai-ichi and will continue work to ensure post-Fukushima safety enhancements are appropriately incorporated into the NRC's existing oversight programs. The NRC research activities related to improving the understanding and modeling of reactor behavior during severe accidents will likewise continue.

This white paper is being provided to support public outreach and its contents should not be interpreted as official agency positions. The paper is currently subject of management and legal reviews and approvals. The NRC staff expects that this topic will be addressed in a future paper from the staff to the Commission.

## Introduction

The NRC's Near-Term Task Force (NTTF) was established shortly after the Fukushima Dai-ichi accident and was directed to conduct a methodical and systematic review of NRC processes and regulations, and provide recommendations to the Commission on whether the agency should make additional improvements to its regulatory program in response to the accident. In SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated July 12, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11186A950), the NTTF provided its recommendations to the Commission. The staff requirements memorandum for SECY-11-0093 (ADAMS Accession No. ML112310021), dated August 19, 2011, directed the staff to recommend a prioritization of the NTTF's recommendations by October 3, 2011.

In SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," dated October 5, 2011 (ADAMS Accession No. ML11272A111), the staff provided the Commission with its proposed prioritization of the NTTF recommendations.

The staff's prioritization approach grouped the recommendations in three tiers. Tier 1 consisted of those NTTF recommendations that the staff determined would lead to the most safety benefit, which should be started without unnecessary delay, and for which sufficient resources and critical skill sets were available. Tier 2 consisted of those NTTF recommendations that could not be initiated in the near-term due to factors that included the need for further technical assessment and alignment, dependence on Tier 1 issues, or availability of critical skill sets. Tier 3 consisted of those NTTF recommendations that require further staff study to support a regulatory action, had an associated shorter-term action that needed to be completed to inform the longer-term action, were dependent on the availability of critical skill sets, or were dependent on the resolution of NTTF Recommendation 1. Most of the Tier 3 items involved studies and assessments.

The NRC staff documented initial project plans for Tier 2 recommendations in SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012 (ADAMS Accession No. ML12039A103). Initial project plans for the Tier 3 recommendations were documented in SECY-12-0095, "Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012 (ADAMS Accession No. ML12208A210).

While the staff's focus to date has been on implementing the higher-priority Tier 1 recommendations, the staff has been working on many of the other recommendations, consistent with the initial project plans. In addition, many of the initial Tier 2 and 3 recommendations have been subsumed into Tier 1 activities (most notably the Mitigation of Beyond-Design-Basis Events rulemaking), and one additional recommendation (evaluation of the need to expedite the transfer of spent fuel to dry cask storage) was completed in May 2014.

**Discussion**

Resolving the open Tier 2 and 3 recommendations is important to timely consideration of Fukushima lessons learned, to support NRC planning and to provide regulatory stability to the industry. As such, the staff began reassessing its initial plans for addressing these recommendations in early 2015 to determine if they could be resolved in light of the progress made on Tier 1 activities, completed regulatory evaluations, and recent Commission decisions. The majority of the open items consist of recommendations to evaluate the need for further regulatory action, rather than a recommendation to take a specific regulatory action (such as issuing an order to require a specific safety enhancement). As such, for each recommendation, the staff's reassessment began with an evaluation of the issue in light of existing requirements and practices to determine if a substantial safety improvement could result from further studies or regulatory action. In conducting these evaluations, the staff considered previous Commission decisions on related matters, information obtained from its work on the Tier 1 recommendations, and the substantial safety enhancement that will be achieved with full implementation of the Tier 1 safety enhancements.

With respect to Tier 1 activities, the safety enhancements being put in place as a result of NRC Orders will substantially improve the already robust protection and mitigation capabilities of U.S. nuclear power plants. These Orders require the development of strategies to mitigate beyond-

design-basis events through maintenance of key safety functions, installation of reliable spent fuel pool instrumentation, and reliable severe-accident-capable hardened containment venting systems for boiling-water reactors with Mark I and II containments. Additional activities, such as seismic and flooding walkdowns and ongoing hazard reevaluations, provide further assurance that nuclear plants will be able to cope with extreme seismic and flooding events. Enhancements to licensees' emergency response capabilities (e.g., in the areas of staffing and communication) are also being put in place. Finally, ongoing rulemaking activities will ensure that actions taken in response to the accident are made generically applicable through incorporation into the regulations, including lessons learned from the implementation of the related orders.

The staff's approach to reassessing the plans for the open Tier 2 and 3 recommendations primarily involved technical evaluations to assess the need for additional regulatory action, considering the existing protection provided under current NRC requirements and new information obtained during completion of Tier 1 activities. This approach is similar to those used by the staff to evaluate lessons learned for facilities other than operating power reactors and to develop the assessment of Fukushima-related recommendations provided by the National Academies of Science in their report on the accident. In some cases, the staff used insights gained from completed analyses, such as the Expedited Transfer of Spent Fuel regulatory analyses and Containment Protection and Release Reduction rulemaking draft regulatory basis, to inform the reviews. The staff prepared the evaluations and the associated resolution plans with the goal of resolving all the remaining Tier 2 and 3 recommendations as efficiently as possible, while maintaining appropriate levels of technical rigor and opportunities for stakeholder involvement.

The staff recognizes that any new regulatory requirement imposed as a result of the open Tier 2 and 3 recommendations must be appropriately justified, as required by Title 10 of the *Code of Federal Regulations*, Section 50.109, "Backfitting." As discussed in the enclosures to this paper and as summarized below, for the majority of the open Tier 2 and 3 recommendations, the staff's evaluation has determined that the NRC's existing regulatory framework and requirements are adequate, and that no further regulatory action or analysis is warranted. However, while the ACRS was provided with an overview of its plans, the staff has not yet fully interacted with ACRS and other stakeholders on the results of its assessments. As such, the staff's resolution plans involve three groups of recommendations:

Group 1: Recommendations that should be closed now. No further assessment or stakeholder interaction is necessary.

Group 2: Recommendations that the staff's initial assessment has concluded should be closed, but for which interaction with ACRS or external stakeholders is warranted prior to finalizing the assessment.

Group 3: Recommendations which the staff believes could benefit from additional assessment and/or documentation, along with ACRS or external stakeholder interaction.

As described in the enclosed plans, for those recommendations in Groups 2 and 3, the staff intends to interact with stakeholders, complete the final assessments, and inform the

Commission of the final results of its evaluation of each open recommendation as soon as practicable and by no later than the end of calendar year 2016. The table below summarizes the way each open Tier 2 and 3 recommendation has been grouped:

| SECY Encl. | Tier | Source of Action | Description of Recommendation | Recommendation Group |
|---|---|---|---|---|
| 1 | 2 | ACRS and Consolidated Appropriations Act 2012 | Evaluation of Other Natural Hazards | Group 3 – complete by December 2016 |
| 2 | 3 | NTTF Rec. 2.2 | Periodic Confirmation of Natural Hazards | Group 3 – complete by December 2016 |
| 3 | 3 | NTTF Rec. 3 | Capabilities to Prevent Seismically-Induced Fires and Floods | Group 1 |
| 4 | 3 | NTTF Recs. 5.2 and 6 | Reliable Vents for Other Containments and Hydrogen Control and Mitigation | Group 2 – complete by March 2016 |
| 5 | 3 | ACRS | Reactor and Containment Enhanced Instrumentation against Beyond-Design-Basis Events | Group 2 – complete by March 2016 |
| 6 | 3 | NRC Staff | Evaluation of Emergency Planning Zone Size and Pre-Staging of Potassium Iodide Beyond 10 Miles | Group 1 |
| 7 | 3 | NTTF Recs. 9, 10, and 11. | Various Emergency Preparedness Activities | Groups 1 and 3 |
| 8 | 3 | NTTF Rec. 12.1 | Enhancements to the Reactor Oversight Process | Group 1 |
| 9 | 3 | NTTF Rec. 12.2 | Enhancements to NRC Staff Training on Severe Accidents and Severe Accident Management Guidelines | Group 1 |

Notwithstanding final resolution of these recommendations, the staff notes that work will continue in some of these areas for many years. For example, the NRC staff will continue to engage with the international community on long-term health studies in the areas around Fukushima Dai-ichi and will continue work to ensure post-Fukushima safety enhancements are appropriately incorporated into the NRC's existing oversight programs. The NRC research activities related to improving the understanding and modeling of reactor behavior during severe accidents will likewise continue.

After issuance of this paper, the NRC staff plans to conduct public meetings on those recommendations it anticipates could benefit from external stakeholder input and incorporate feedback, as appropriate. The staff has interacted with the ACRS to provide an overview of the initial assessments and resolution plans. A meeting with the ACRS's Fukushima subcommittee was held on October 6, 2015, and an ACRS full committee meeting is scheduled for November 5, 2015. The staff will also incorporate feedback from ACRS and other stakeholders

in developing a final assessment of each of those recommendations.

**Milestones**

- JLD public Steering Committee meeting with industry – October 20, 2015
- SECY paper provided to the Commission – October 30, 2015
- ACRS Full Committee Meeting – November 5/6, 2015

**Enclosure 1: Proposed Resolution Plan for Tier 2 Recommendation on Other Natural Hazards**

**Background**

As directed by the staff requirements memorandum to SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated August 19, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112310021), the staff sought to identify additional recommendations related to lessons learned from the Fukushima Dai-ichi event, beyond those identified in the Near-Term Task Force report. Many additional recommendations were received from the U.S. Nuclear Regulatory Commission (NRC) staff and external stakeholders, including the Office of Science and Technology Policy, Congress, international counterparts, other Federal and State agencies, nongovernmental organizations, the public, and the nuclear industry. These issues were raised in a variety of forums, including the staff's August 31, 2011, public meeting and a September 9, 2011, Commission meeting.

As part of that initiative and in response to comments from the Advisory Committee on Reactor Safeguards (ACRS) and specific language included in the Consolidated Appropriations Act, 2012 (Public Law (Pub. L.) 112-74, dated December 23, 2011), the NRC staff identified an action regarding reevaluations of natural external hazards other than seismic and flooding hazards. In SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated March 9, 2012 (ADAMS Accession No. ML12039A103), this was prioritized as a Tier 2 activity because of the lack of availability of the critical skill sets for both the NRC staff and external stakeholders and because the NRC staff considered the seismic and flooding reevaluations to be of higher priority.

Enclosure 3 to SECY-12-0025 detailed the initial program plan for this recommendation. That plan called for the staff to follow the same process used for the Tier 1 seismic and flooding reevaluations (i.e., issue a request for information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(f)).

Section 402 of Pub. L. 112-074 requires a reevaluation of licensees' design bases for external hazards and expands the scope to include other external events, as described below:

> The Nuclear Regulatory Commission shall 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. 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.

Subsequently, the NRC's Office of Congressional Affairs, during interactions with House and Senate Appropriations Committee staff, clarified that the intent of Pub. L. 112-074 was for the NRC to include natural external hazards in the scope of its review, not man-made hazards. Because man-made hazards do not have a direct nexus to the Fukushima Dai-ichi accident, the staff concluded that they should be treated outside the scope of Fukushima lesson-learned activities. As such, the NRC staff submitted the consideration of man-made hazards to the NRC's Generic Issue (GI) program by memorandum dated September 9, 2013 (ADAMS Accession No. ML12328A180). By memorandum dated January 17, 2014 (ADAMS Accession No. ML13298A782), the NRC staff concluded that the proposed GI does not satisfy at least three criteria for acceptance as a GI. Therefore, the NRC staff did not undertake possible regulatory requirements or information collection related to man-made hazards and will continue to address issues in that area as they arise on a case-by-case basis, as has been the NRC's historical practice.

**Current Status**

The NRC staff has reviewed a variety of domestic and international documents related to external hazards. The staff concluded that the most prevalent natural hazards, beyond seismic and flooding, are extreme winds, extreme temperatures, drought and other low-water conditions, and winter precipitation that results in snow and ice loading on structures. The current regulatory framework requires that all U.S. nuclear sites be evaluated for these hazards when initially licensed. As required by 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," licensees shall demonstrate that their safety-related structures, systems, and components are designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions, giving 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.

To complete the Tier 2 activity and comply with the requirements of Section 402 of the Consolidated Appropriations Act, the NRC staff is proposing to evaluate the Tier 2 external hazards utilizing existing information and processes, and assess the need for further regulatory actions. This will be done considering previously submitted licensee information on external hazards, such as: information provided in the licensee's integrated plans required by Order EA 12 049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012 (ADAMS Accession No. ML12054A735); licensee information on the criteria used for its plant's design and licensing basis; information from recent NRC activities related to the prevalent hazards (for example, Regulatory Issue Summary 15-06, "Tornado Missile Protection," dated June 6, 2015); and recent GI program reviews.

**Discussion**

Seismic and flooding hazards were given priority as Tier 1 activities during the NRC's review of Fukushima lessons learned because of the risk these hazards pose to operating plants and due, in part, to significant advancements in the state of knowledge and the state of analysis in these areas since the operating plants were sited and licensed.

The state of knowledge and the state of analysis has also advanced for other natural hazards, such as extreme winds. In some cases our improved understanding of the hazard has led the staff to determine the hazard level previously considered was more conservative than that required today. For example, many of the current operating plants used guidance that relied on the Fujita scale to relate the degree of damage to maximum tornado wind speed. Current guidance relies on the Enhanced Fujita scale, which reduces the maximum tornado wind speed for a given damage state, meaning many currently operating plants used higher tornado wind speeds to design the plant than would be required today. However, improved understanding and enhanced models have also indicated that for some sites, missiles generated from hurricane winds (which are often lower than design basis tornado wind speed) may produce the most intense missiles. Given the information now available to the staff as a result of the ongoing mitigating strategies work, and recognizing that there may be a significant population of plants where the current design basis hazard maybe more conservative than that developed using modern methods, the staff proposes a screening approach that would focus resources on those sites that provide the most opportunity to gain a safety benefit.

As part of its review of this issue, the NRC staff has considered how other natural external hazards are being addressed within the requirements for mitigating strategies for beyond-design-basis external events. The guidance in NRC-approved Nuclear Energy Institute (NEI) 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," describes a process for licensees to determine which external hazards should be addressed within the mitigating strategies developed for each site. The staff notes that a safety benefit has been achieved in the near term for the Tier 2 hazards as well as seismic and flooding, because external events associated with these hazards have been considered in the implementation of Order EA 12-049 and are being considered in the proposed rule for Mitigation of Beyond Design Basis Events (MBDBE). Nevertheless, consistent with the Consolidated Appropriations Act of 2012, the staff believes that additional review should be performed to determine if changes in the hazard warrant other actions, beyond those associated with Order EA-12-049 and the MBDBE proposed rule, to ensure public health and safety against external hazards other than seismic and flooding.

The staff notes that the safety benefit achieved through Order EA-12-049 should be factored into an evaluation of potential regulatory requirements to determine whether additional changes could be justified when evaluated against the criteria in 10 CFR 50.109 for the backfitting of operating reactors.

The NRC staff has divided this review into the following four tasks:

1. Define nature hazards other than seismic and flooding to determine those hazards that should be reviewed generically (complete).

Documents reviewed included the following:

- Electric Power Research Institute 1022997, “Identification of External Hazards for Analysis in Probabilistic Risk Assessment,” dated December 7, 2011
- American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009 Appendix 6-A, “Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications”
- International Atomic Energy Agency TECDOC-1341, “Extreme External Events in the Design and Assessment of Nuclear Power Plants,” issued March 2003
- Nuclear Energy Agency (NEA)/CSNI/R(2009)4, “Probabilistic Safety Analysis (PSA) of Other External Events Than Earthquake,” dated May 5, 2009
- NUREG/CR-5042 “Evaluation of External Hazards to Nuclear Power Plants in the United States,” issued December 1987
- NUREG-0800 “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition”
- Other international and domestic references

Using previous analysis and engineering judgment, the other external hazards to be evaluated are wind and missile loads from tornadoes and hurricanes, 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 (postaccident), and meteorological conditions related to the maximum evaporation and drift loss and minimum water cooling for the ultimate heat sink design.

2. Determine and apply screening criteria to appropriately exclude certain natural hazards from further generic evaluations, or exclude some licensees from considering certain hazards. Possible screening criteria may include:

    - Conservatism due to adequate design safety margins.
    - Operational limits provided in technical specifications.
    - Low frequency of occurrence/low risk.
    - Warning time available to allow measure to be taken to avoid an accident from occurring.

    This process will consider, among other things, whether external hazards should be eliminated from consideration because they are addressed by existing requirements (e.g., temperatures affecting ultimate heat sinks) or common industry preparations for severe weather such that it is unlikely the hazard will cause an accident. The focus of discussions related to other external hazards has been wind events, and primarily tornados, because of limited time for licensees to prepare for the event. However, some plants may have designed to winds speeds and missiles that are more severe than our current guidance requires.

3. Perform a technical evaluation to assess the need for additional actions if the hazard or licensee were not screened out generically in Task 2.

If a site-specific evaluation is needed:

- 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 or mitigating strategies in place to address the impact of the hazard.).
- Request that licensees reevaluate site-specific external natural hazards.

The NRC guidance for determining if requests for information from licensees are warranted is provided in NRC Management Directive (MD) 8.4, "Management of Facility-Specific Backfitting and Information Collection," dated October 28, 2004. Regarding the evaluation of other external hazards, the guidance in NEI 12-06 includes a site-specific screening of external events for each site and the NRC staff has reviewed the results of the screenings in overall integrated plans submitted by licensees. The staff's work on Task 3 will consider the work already completed in the development of mitigating strategies, along with previous staff assessments, such as the Individual Plant Evaluation of External Events. The staff notes that some events, such as high temperatures, may already be addressed in by specific regulatory requirements (e.g., in a given facility's technical specifications). The staff also will consider this factor in its assessment. The staff's plans call for considering the results of previous assessments in its evaluation of this issue, preparing more detailed documentation of the technical evaluation, and scheduling additional public interactions for this activity.

4. Determine if additional actions are needed, such as the following:

- Evaluate the results from Task 3, including actions taken or planned by the licensee and determine if additional action is needed. Any further regulatory actions will require a formal and systematic review to ensure that changes are properly justified and suitably defined as required in 10 CFR 50.109.
- Issue generic communications per MD 8.18, "NRC Generic Communications Program," dated March 5, 2009.

The NRC guidance for evaluating the possible imposition of additional requirements on licensees for operating nuclear power plants is also provided in NRC Management Directive 8.4. As part of Task 4, the staff would use the information developed to determine if a facility-specific backfit it necessary, based on the guidance in Management Directive 8.4 and 10 CFR 50.109. As noted above, the staff would also consider other regulatory options, such as issuance of a generic communication, depending on the results of its assessment.

**Stakeholder Interactions**

During a meeting held on October 6, 2015, the NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff's plans to resolve the open Tier 2 and 3 recommendations. A similar meeting is planned with the ACRS full committee on November 5, 2015.

Before completing its review of this recommendation, the NRC staff plans to discuss its results with external stakeholders, including the industry and members of the public. The staff will then conduct a focused briefing on this issue with ACRS, if appropriate, before providing its final assessment to the Commission.

**Conclusion**

The NRC staff will further evaluate the risk to U.S. nuclear sites from other natural external hazards and determine if associated risks warrant regulatory action. The staff will provide the results of this evaluation and the justification supporting resolution of this issue to the Commission by the end of 2016. The final assessment will also consider the outcome of interactions with the ACRS and external stakeholders.

**Enclosure 2: Proposed Resolution Plan for Tier 3 Recommendation 2.2, Periodic Reconfirmation of Natural Hazards**

**Background**

As described in SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated December 23, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11186A950), the U.S. Nuclear Regulatory Commission's (NRC's) Near-Term Task Force (NTTF) Recommendation 2.2 suggested that the NRC initiate a rulemaking to require licensees to confirm seismic and flooding hazards every 10 years and address any new and significant information. In SECY-11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," dated October 5, 2011 (ADAMS Accession No. ML11272A111), the staff prioritized Recommendation 2.2 as a Tier 3 item because it is associated with Recommendation 2.1, a Tier 1 item requiring licensees to reevaluate the flooding and seismic hazards using present-day methodologies and guidance. In the staff requirements memorandum for SECY-11-0137, dated December 15, 2011 (ADAMS Accession No. ML113490055), the Commission agreed with the Tier 3 prioritization of Recommendation 2.2.

The initial program plan for this recommendation was detailed in SECY-12-0095, Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012, Enclosure 3, "Program Plans for Tier 3 Recommendations" (ADAMS Accession No. ML12208A210). The original program plan defined the initial prerulemaking activities necessary to position the agency for a future rulemaking to implement NTTF Recommendation 2.2. In the initial program plan, the staff indicated that as it gains experience from the implementation of Recommendation 2.1 and knowledge from the prerulemaking activities, it would develop a complete rulemaking plan for Recommendation 2.2.

Section 402 of the Consolidated Appropriations Act, 2012, (Pub. L. 112-074), requires a reevaluation of the licensees' design bases for external hazards and expands the scope to include other external events, as described below:

> The Nuclear Regulatory Commission shall 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. 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.

In SECY-12-0095, the staff discussed that this language indicates that other external hazards, such as those caused by meteorological effects, should be included in the periodic updates that would be required once Recommendation 2.2 is implemented.

**Current Status**

To date the NRC staff has made significant progress on the Tier 1 seismic and flooding reevaluations. These reviews have provided the staff with important insight on the need for a rule to require licensees to periodically confirm their external hazards. It is the staff's view that the NRC's current regulatory framework is sufficient to effectively consider the implications of new external hazard information on plant safety. While the staff's assessment did not identify the need for a new rule, the staff has determined that enhancing its current processes would improve the staff's efficiency in identifying and assessing new information related to external hazards. These enhancements would allow the staff to be more proactive in identifying potentially meaningful changes in our understanding of how external hazards effect plant safety, and more efficient in identifying any necessary changes. To provide context, the remainder of this section summarizes how external hazards are assessed under our current regulatory framework.

The NRC has long recognized the importance of protection of NRC-licensed facilities from natural phenomena as a means to prevent core damage and to ensure containment and spent fuel pool integrity. Several requirements were established addressing natural phenomena in 1971 with General Design Criteria (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." GDC 2 requires, in part, that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena such as earthquakes, floods, tsunami, and seiches, without loss of capability to perform their safety functions. GDC 2 also requires that design bases for these SSCs reflect (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding region, with sufficient margin for the limited accuracy and quantity of the historical data and the period of time in which the data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed. Through its initial licensing process, the NRC's regulatory structure ensures that plants are thoroughly and comprehensively reviewed prior to allowing operation to begin.

Following initial licensing, the NRC has historically evaluated external hazard information as it has been identified and taken actions to update guidance or to impose regulations, as needed, consistent with the regulatory processes in 10 CFR 50.54(f) and 10 CFR 50.109. Further, through the inspection and oversight program, the NRC monitors plant performance and operating experience on a daily basis. There are a variety of ways that the NRC maintains cognizance of developments in the area of natural hazards. For example:

- Operating experience: There have been a number of natural events that have affected nuclear sites around the world. These events include tsunamis, flooding, high winds, and seismic events. In some cases, the plant's design basis was exceeded during these events. The NRC and industry routinely conduct investigations to identify the lessons learned from these events. In addition to the Fukushima event, recent examples have included the 2004 Sumatran tsunami; ground motions experienced at Japan's Kashiwazaki-Kariwa site during a large earthquake in 2007; the 2011 Mineral, Virginia earthquake; and recent flooding at Fort Calhoun.

- Research: The NRC and industry (through the Electric Power Research Institute) have worked cooperatively on projects to assess the impact on plant safety of the latest understanding of natural phenomenon hazards, analytical advances, and evaluation tools. In addition, joint projects with the other U.S. Federal Agencies, such as the Department of Energy (DOE), United States Geological Survey, and the National Oceanic and Atmospheric Administration, have enabled the NRC to monitor and cooperatively assess new information.

The evaluations completed in response to information obtained from these sources has led to new requirements, updated regulatory guidance, generic communications, and plant-specific actions to address identified issues.

NRC's regulatory framework provides for licensee review of new hazard information and, as necessary, consideration and resolution of new information through one or more regulatory processes, including the following:

- Formal corrective action programs under 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action."

- Operability determinations as described in NRC Regulatory Issue Summary 2005-20, "Revision to NRC Inspection Manual Part 9900 Technical Guidance, 'Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety,'" Revision 1, dated April 16, 2008.

- 10 CFR 50.59, "Changes, Tests and Experiments."

- 10 CFR 50.9(b), "Completeness and Accuracy of Information."

- RG 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, issued February 1978 (ADAMS Accession No. ML003739995).

- RG 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," issued September 1999 (ADAMS Accession No. ML14346A207).

A number of studies conducted after the Fukushima accident included recommendations that further emphasize the importance of assessing new information. For example, Finding 3.1 of the National Academies of Sciences report, "Lessons Learned from the Fukushima Nuclear Accident for Improving Safety of U.S. Nuclear Plants," states: "The overarching lesson learned from the Fukushima Dai-ichi accident is that nuclear plant licensees and their regulators must actively seek out and act on new information about hazards that have the potential to affect the safety of nuclear plants."

**Discussion**

As discussed above, the NRC and its licensees continually evaluate new information as it

becomes available to assess its potential impact on risk and overall plant safety. However, while the staff finds that current practices are adequate for ensuring plant safety, the staff has also identified that there is benefit to enhancement of the current processes to facilitate a more proactive and systematic assessment of new information related to external hazards, consistent with the recommendations of the NTTF and the National Academies of Science.

The NRC's current practice generally involves initiating a hazard reassessment either after the occurrence of a major event that challenges a plant's design basis or after receipt of information determined to have the potential to significantly impact plant safety. The staff's suggested approach takes into account the strengths of the current NRC processes and strives to achieve additional efficiencies through implementing an approach that the staff believes will result in a more predictable, stable process that will not place undue burden on NRC licensees and will lead to greater efficiencies in the current process for identifying and evaluating new information.

The NRC staff has examined possible alternatives for adopting the approach described above, including the use of rulemaking and the development of an internal program. As discussed below, as part of its assessment, the staff has concluded that rulemaking is not a viable alternative for addressing this recommendation. Instead, the staff has found that supplementing the current regulatory framework with a formal internal process to more systematically review information related to natural hazards is the preferred alternative.

*Viability of Rulemaking to Address Recommendation 2.2*

The original intent of NTTF Recommendation 2.2 was to initiate a rulemaking to require licensees to confirm seismic and flooding hazards every 10 years and address any new and significant information. As previously described, this confirmation would also include other natural hazards (but would exclude man-made hazards).

Under the initial project plan for Recommendation 2.2, rulemaking would depend on insights gained from the ongoing seismic and flooding reevaluations. When sufficient insights are gained from the seismic and flooding reevaluations, the NRC staff planned to start the rulemaking process. However, no significant work has been initiated on that rulemaking to date.

The current regulatory framework is described above. As discussed above, if a new natural external hazard or a significant change is discovered which may impact the previously determined design basis hazard analysis, the NRC has existing regulatory processes to ensure licensees address the new information that could affect the plant. These existing processes provide reasonable assurance that any new hazard information that is discovered will be properly evaluated. However, these processes have not always resulted in efficient, predictable, and transparent resolution of new information related to external hazards.

Through an extensive inspection regime, the NRC monitors plant performance and operating experience on a daily basis. Results from these inspections are coupled with objective performance indicator data, and then assessed in a comprehensive manner by NRC senior management and staff at a mid-year and end-of-year assessment to verify that plants continue to operate safely. Although the NRC does not require its licensees to summarize performance with a recurring 10-year submittal to the regulator, as is done with a PSR, the NRC staff's

assessment is that its continuous inspection and assessment approach ensures that safety issues are promptly identified and evaluated systematically.

Given these considerations, the NRC staff believes that any changes to be made through a proposed rulemaking for Recommendation 2.2 are not necessary and would likely not be justified if evaluated against 10 CFR 50.109. Therefore, the staff concludes that imposing a regulatory requirement on licensees to perform the periodic assessments is not necessary, and therefore not a viable approach for closing out Recommendation 2.2.

*Staff's Proposed Approach – Develop an Internal Program*

As discussed above, current NRC processes assess information when it is identified, but there is no existing NRC process which actively seeks to determine if there is new hazard information available and assess the impact of that new information on the safety of NRC-licensed facilities. As such, when new information is identified, there is the potential that the information could be evaluated in isolation, rather than methodically evaluating the cumulative effect of new data, models, and methods that have accrued over time. In addition, because existing hazard models are not routinely updated with new information, additional resources and time are required to update those methods and models when new information is identified. This may lead to challenges with regulatory predictability and efficiency, particularly if there is a need to evaluate new information quickly following a significant event, such as with the Fukushima Dai-ichi accident.

In order to address these issues, as part of the staff's proposed approach for resolving Recommendation 2.2, the NRC staff proposes to leverage and enhance existing NRC processes and programs to ensure that information related to external hazards is proactively and routinely evaluated in a systematic manner. The staff would continue to assess the impact of any new and significant information on the safety of NRC-licensed facilities at the time that information is known to the agency, but the staff also proposes to enhance existing processes to ensure that all new hazard information has been proactively and routine aggregated by the NRC. These procedures will also provide guidance for determining the significance of the totality of all the new hazard information. The enhanced internal process will leverage and augment existing programs and agreements with domestic and international organizations. The details of this enhancement, including which process will be used to obtain information and the process for screening the information to determine its significance, are still under development and would be completed by the end of calendar year 2016.

To the degree possible, the NRC will partner with other Federal agencies and industry to systematically and periodically evaluate new data, models, and methods, and assess their impact on currently-licensed facilities. As other Federal agencies are developing processes for routinely updating external hazard information, enhancement of current NRC processes will permit the NRC to have influence and access to this information. Partnering with other Federal agencies will increase consistency across the government and permit an overall cost saving to the NRC and Federal government. Should the periodic assessment identify information that might potentially impact plant licensing bases, the new information would be input to the appropriate program for consideration (e.g., the Generic Issues Program). If the licensees are required to evaluate an external study under any of the regulatory processes discussed above, the procedure will provide a path forward for appropriate licensee response. Part of this

process will provide for routinely updating of the NRC models so that the staff is prepared to assess the significance of new information received. The institutionalization of this procedure will benefit from the experience gained through various activities associated with Recommendation 2.1. For example, the staff was able to rapidly assess the Recommendation 2.1 seismic hazard reevaluations for all of the central and eastern United States reactor sites due to the availability of recently developed regional seismic models, which were jointly developed by the NRC, industry, and the DOE for new reactor siting. The fact that these models were developed in anticipation of new reactor applications was fortuitous, because flooding and seismic hazard models are not currently routinely updated. Based on this experience, the staff believes it is important to maintain these models in a cooperative fashion with industry and other Federal agencies, such that an expedient and timely hazard reevaluation of NRC-licensed facilities can be implemented. Most significantly, this approach will provide a more systematic and proactive approach, consistent with Finding 3.1 of the National Academies of Sciences report. It will also further enhance NRC partnerships with other Federal agencies and external stakeholders to achieve the common goal.

Processes for incorporating new information into NRC regulatory processes and screening new information for significance already largely exist, and the staff would build off of these current processes to ensure the routine aggregation of new external hazards information received over time be institutionalized at the NRC in order to gain efficiency and minimize resource implications. While the staff notes that while there will be some resources implications associated with this approach, both in initial development of the program and on a recurring basis, the following benefits outweigh the associated costs:

- New hazard information would be proactively sought and addressed.
- A systematic process would be in place that will allow NRC staff to identify new hazard information and conduct the appropriate evaluations.
- To the extent possible, the new program will leverage existing NRC processes to achieve the objectives in an efficient manner.
- There will be no burden on licensees unless an actual safety-significant change in hazard information is identified by the NRC.
- The process will avoid ad hoc reactive response with the resultant strain on resources, improving timeliness and efficiency.
- The systematic nature of the process will ensure transparency and consistency in dealing with new hazard information evaluations.
- The approach will address concerns expressed in the NTTF report by increasing confidence that revised estimates of external hazards will be periodically assessed.
- The process will leverage existing regulatory processes, existing research programs, and cooperation with other Federal agencies.
- The process will strengthen the NRC's predictability and consistency as a regulator.
- The process will promote joint NRC and industry partnerships.

In summary, the staff's assessment has concluded that Recommendation 2.2 should be addressed through the enhancement of internal processes that establishes a routine, proactive, and systematic program for identifying and evaluating new information related to external hazards.

**Stakeholder Interactions**

The NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015.

The staff intends to discuss the results of this review and development of the program discussed above with industry and public stakeholders. The NRC staff will also conduct a focused briefing for the ACRS, if desired, as the process is developed. The NRC staff will inform the Commission if the recommendation discussed above changes based on additional analysis or as a result of stakeholder interactions.

**Conclusion**

In summary, the NRC staff has evaluated the need to take action to routinely evaluate new information related to natural external hazards and address any new and significant information, as recommended by the NTTF in Recommendation 2.2. The staff has concluded that the use of rulemaking to address this recommendation, as originally discussed in Recommendation 2.2, is not a viable approach. Rather, the staff proposes to develop a method to leverage and enhance existing NRC processes and programs to ensure that information related to external hazards is proactively and routinely evaluated in a systematic manner. The staff anticipates completing the enhanced process and programs before the end of Calendar Year 2016.

**Enclosure 3: Proposed Resolution Plan for Tier 3 Recommendation 3 – Potential Enhancements to the Capability to Prevent or Mitigate Seismically-Induced Fires and Floods**

**Background**

As described in SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated December 23, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11186A950), the U.S. Nuclear Regulatory Commission's (NRC's) Near-Term Task Force (NTTF) identified that seismically-induced fires have the potential to cause multiple failures of plant safety equipment and induce separate fires in multiple locations at the site. It has also been recognized that events such as pipe ruptures (and subsequent flooding) could cause such problems in multiple locations simultaneously. Additionally, seismic events could degrade the capability of plant safety equipment intended to mitigate the effects of fires and floods. Although these issues have been examined to a certain extent in previous agency evaluations, Recommendation 3 of the NTTF's report concluded that the staff should evaluate potential enhancements to the capability to prevent or mitigate seismically-induced fires and floods (SIFFs).

In SECY-11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011 (ADAMS Accession No. ML11272A111), the staff prioritized this recommendation as a Tier 3 activity since it required further staff study to support a regulatory action. In the staff requirements memorandum (SRM) to SECY-11-0137, the Commission agreed with the Tier 3 prioritization of Recommendation 3, but directed the staff to initiate the development of a probabilistic risk assessment (PRA) methodology to evaluate potential enhancements to the capability to prevent or mitigate SIFFs as part of Tier 1 activities. Tier 1 activities are those NTTF recommendations that should be worked on without unnecessary delay.

In SECY-12-0095, "Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012 (ADAMS Accession No. ML12208A210), the staff provided a program plan to address the Tier 1 and Tier 3 portions of Recommendation 3. In that program plan, the staff developed a multistep process to gather additional information to inform future determinations on the need for regulatory action to address this recommendation.

In SECY-15-0059, "Seventh 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated April 9, 2015 (ADAMS Accession No. ML15069A444), the staff provided the Commission with the most recent update on the status of work on this recommendation. In this update, the staff described the status of efforts on the Tier 1 activity to develop a PRA methodology and indicated that its goal is to complete a feasibility study by December 2015. The staff also provided a general estimate of the timeline for a further update of the Tier 3 evaluation plan, in consideration of information obtained from Tier 1 activities and PRA development activities.

Separate from the work described above on Recommendation 3, the Advisory Committee on Reactor Safeguards (ACRS) provided a related recommendation during its review of the NTTF report. In its letter entitled, "Initial ACRS Review of: (1) The NRC Near-Term Task Force Report on Fukushima and (2) Staff's Recommended Actions to be Taken Without Delay," dated October 13, 2011 (ADAMS Accession No. ML11284A136), the ACRS noted that integration of onsite emergency response capabilities envisioned by Recommendation 8 (related to strengthening and integrating onsite emergency response capabilities, such as emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and extreme damage mitigation guidelines (EDMGs)) should be expanded to include fire response procedures. In SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012 (ADAMS Accession No. ML12039A103), the staff conducted an initial evaluation of the ACRS recommendation to integrate the fire response procedures into the licensee's onsite emergency response capabilities and determined that this recommendation would be best considered with the agency's Tier 3 actions associated with Recommendation 3. The staff provides a further analysis of the ACRS recommendation in this document.

**Current Status**

The staff has completed its assessment of this recommendation and has determined that additional enhancements to existing capabilities to prevent or mitigate SIFFs would not be warranted. The rationale for this assessment is discussed in further detail below.

With respect to the Tier 1 portion of this recommendation, the staff is working closely with Brookhaven National Laboratory (BNL) to finalize a feasibility study on the development of a PRA methodology for SIFFs. BNL has completed a draft version of the feasibility study, which the staff is currently reviewing for technical accuracy. The staff has also shared the draft study with subject matter experts and has solicited comments. Once all of the comments are received, BNL will appropriately address them and finalize the document. The staff expects to have this report finalized by December 2015.

The staff will continue participating with standard development organizations and updating regulatory guidance documents on this subject as part of its routine activities. For example, the staff continues to work with the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) Joint Committee on Nuclear Risk Management (JCNRM) to leverage external stakeholders' expertise and to better focus future method-development efforts. Following JCNRM approval of the incorporation of crosscutting issues in the ASME/ANS PRA standard, including concurrent initiating events such as SIFFs, implementation guidance was supplied to the PRA writing groups associated with affected parts of the standard. The staff will continue working with ASME and ANS to support the development of detailed standards requirements in this area. The staff plans to manage this activity through normal agency processes outside of Fukushima lessons-learned initiatives.

**Discussion**

It has been previously recognized that a seismic event at a nuclear power plant (NPP) could cause additional adverse impacts, such as fires or floods, which could result in multiple failures of safety-related components in different locations and potentially degrade the capability of plant

components intended to mitigate the effects of fires and floods. The staff identified SIFFs as a potential risk contributor during the resolution of Generic Safety Issue 172, also known as the multiple system responses program. In NUREG/CR-5420, "Multiple System Responses Program – Identification of Concerns Related to a Number of Specific Regulatory Activities," dated October 1989 (ADAMS Accession No. ML072420007), the staff determined it needed additional information to assess the safety significance of the issue and it proceeded with issuing a generic letter pursuant to Paragraph 50.54(f) of Title 10 of the *Code of Federal Regulations* (10 CFR).

The staff issued Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," dated June 28, 1991 (ADAMS Accession No. ML031150485), requesting licensees to evaluate plant-specific vulnerabilities to severe accidents and report the results to the Commission. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," issued June 1991 (ADAMS Accession No. ML063550238), provided guidance that SIFFs were to be considered as part of the IPEEE. The results of the IPEEE program were documented in NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program – Final Report," dated April 2002 (ADAMS Accession Nos. ML021270070 and ML021270122). No further regulatory action was pursued by the NRC as a result of these activities. As part of the NTTF reviews, the staff determined that some licensees did consider the potential for SIFF events in their IPEEE studies, but they did not identify any vulnerabilities from this effort. However, the staff found that the level of effort, scope, and detail considered for SIFFs during IPEEE studies varied significantly among licensees. Because this issue was only examined to a limited extent as part of the IPEEE program, the NTTF recommended that the staff reevaluate this issue in light of updated analysis techniques and lessons learned from operating events.

To determine if regulatory action is needed to require enhancements to existing capabilities to prevent or mitigate SIFFs, the staff developed a two phased approach, which was outlined in SECY-12-0095. The purpose of the first phase was to evaluate the feasibility of developing a PRA methodology suitable for identifying potential enhancements to the capability to prevent or mitigate SIFFs. During the first phase, the staff also monitored the progress of other related NTTF activities relevant to SIFFs, such as Recommendations 2.1, 2.3, and 4.2. The purpose of the second phase is to evaluate whether regulatory action is needed to prevent or mitigate a SIFF. As outlined in the original plan, and as directed in the SRM for SECY-11-0137, the staff had intended to use the results of the Phase 1 activity to conduct the Phase 2 evaluation to determine whether regulatory action is needed. However, as there has been significant progress and knowledge gained from related activities and because of challenges associated with the development of a PRA methodology, the staff has decided to proceed with the Phase 2 evaluation in this paper. The status of both of these phases is described below.

Feasibility Study

As directed by the Commission in SRM-SECY-11-0137, the staff initiated the development of a PRA methodology to evaluate potential enhancements to the capability to prevent or mitigate SIFFs. The NRC defines a PRA as a systematic method for assessing risk by determining what can go wrong, how likely it is, and what its consequences might be. By developing a PRA for SIFFs, the staff could gather additional risk information to inform future decisions on whether

regulatory action is needed in this area. Although there has been much research and analysis on developing PRAs in general, as acknowledged by the staff in SECY-12-0025, there are no current state-of-practice PRA methods capable of supporting a quantitative assessment of SIFFs. Since the staff could not easily use an existing PRA tool to evaluate SIFFs, a contract was initiated with BNL to support a scoping study on the technical feasibility of developing a method (or a graded approach) for assessing SIFFs within a risk-informed framework.

BNL was tasked with generating a report with sufficient information about the feasibility of a PRA approach to assess risk from SIFFs, such that the information could be used to make informed decisions about the appropriate next steps to take. To gather the information needed to inform the feasibility of developing PRA tools, BNL and the staff conducted the following activities: surveyed available literature on SIFFs; organized a workshop on SIFFs that identified the challenges and potential approaches related to modeling the risk from these hazards (a summary of the workshop is available at ADAMS Accession No. ML14022A249); and solicited expert advice through three questionnaires to develop qualitative screening tools of these hazards. BNL recently completed a draft version of this feasibility study and proposed the development of a graded screening approach for SIFFs.

The feasibility study identified a number of key issues associated with the development of qualitative or quantitative PRA methods for SIFFs. Through the workshops and expert consultation, a number of unresolved technical issues have been identified that will need to be evaluated further in order to develop a PRA methodology. The staff expects that in order to fully develop the PRA methodology, a pilot application will be needed to test and refine the method. The staff concluded, based on the feasibility study and related activities, that finalization of the PRA methodology for SIFFs will require additional time and resources. Furthermore, in addition to completing a pilot application, PRA assessment of SIFFs would require plant-specific seismic, fire, and flooding PRA models and the development of component seismic fragility data for fire and flooding. However, based on the information gathered thus far, the staff did not identify any information that indicates that SIFFs represents a significant safety issue. Therefore, the staff believes that the potential benefits of fully developing and piloting a PRA approach for SIFFs will not justify the costs of continuing this effort. Therefore, the staff expects that the issuance of this report in December 2015 will complete the staff's work on the development of a PRA method to evaluate the risk of SIFFs.

Evaluation of SIFFs

As discussed above, the staff determined that there are no current-state-of-practice PRA methods for evaluating the risk from SIFFs. However, based on insights from the SIFFs PRA development activities conducted, the staff believes that the risk contribution from SIFFs is relatively small compared to that from seismic events alone. This conclusion is based on the robustness of plant mitigation capabilities, the existence of layers of protection, and response mechanisms that already exist or will be in place as a result of related NTTF activities. Additionally, it is expected that any activities to further prevent or mitigate SIFFs would only apply to a portion of the seismic hazard spectrum due to the available plant margins for lower-level seismic events and the difficulty in identifying cost-effective enhancements for higher-magnitude seismic events.

Separate from the development of a PRA method to assess the risks from SIFFs, the staff conducted a deterministic evaluation of SIFFs to decide whether regulatory action is needed on this issue and to complete the Tier 3 portion of Recommendation 3. The staff is conducting this deterministic evaluation due to the lack of readily available probabilistic risk information and the availability of sufficient information from related activities. To conduct this evaluation, the staff considered a variety of topics relevant to this issue. These topics include a review of recent regulatory activities pertaining to seismic, fire, and flooding events, as well as operating experience involving SIFFs, and actions taken in response to the Fukushima accident. Through the review of these materials, the staff has determined that regulatory action is not needed to require enhancements to the existing capabilities of NPPs to prevent or mitigate SIFFs.

The staff considered the following factors in reaching its conclusion that no further action is needed on this issue:

- Robustness of fire protection programs, including fire protection defense-in-depth, under existing NRC requirements.

- Ability of NPP systems to prevent or mitigate internal flooding events under the NRC's existing regulatory requirements.

- Seismic walkdowns conducted as part of NTTF Recommendation 2.3 have identified and corrected potential vulnerabilities from SIFF events.

- Implementation of a robust mitigation capability as part of NRC Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated August 29, 2012, and related rulemaking activities.

- A review of domestic and international operating experience, which did not identified any vulnerabilities that would warrant further regulatory actions.

These factors are discussed in more detail below.

One of the primary mechanisms of preventing seismically-induced fires from affecting reactor safety systems is the NRC's requirement for all NPP licensees to maintain a robust fire protection program. Through the concept of fire protection defense-in-depth, the NRC uses multiple layers of defense to protect the health and safety of the public from fires at nuclear power plants by (1) preventing fires, (2) detecting and suppressing fires that do occur, and (3) protecting the safe shutdown capability of the reactor in the event an uncontrolled fire does occur. Examples of features employed at NPPs for fire protection include fire barriers, fire detection systems, and fire suppression systems. In accordance with 10 CFR 50.48(a), each plant is required to have a fire protection plan outlining the fire protection program, installed fire protection systems, and the means to ensure the reactor can be safely shutdown in the event of a fire. The NRC regularly inspects how plants achieve and maintain the reactor's safe shutdown capability in the event of a fire. From early fire protection guidelines, such as Branch Technical Position APCSB 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976," dated August 23, 1976 (ADAMS Accession No. ML07066048),

to current guidance as discussed in Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants," issued October 2009 (ADAMS Accession No. ML092580550), the NRC has an expectation that fire protection capability would function following a safe-shutdown earthquake. If a seismically-induced fire were to be initiated, the NRC has confidence that the NPP's fire protection program would protect the safety of the reactor.

Additionally, as an alternative to the requirements of 10 CFR 50.48(b), the NRC has allowed NPPs the opportunity to adopt, on a voluntary basis, National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants" (accessible at http://www.nfpa.org). This initiative to transition to a risk-informed, performance-based fire protection program under 10 CFR 50.48(c) is part of an NRC effort to incorporate risk information into the agency's regulations and enhance safety. NFPA 805 describes the fundamental fire protection program elements and minimum design requirements for fire protection systems and features to satisfy the performance criteria. NFPA 805 focuses on reactor safety-oriented fire protection, adds appropriate flexibility, and provides a more detailed evaluation of safe-shutdown conditions in the event of a fire. As of August 2015, a total of 20 NPP units have transitioned to NFPA 805, with an additional 26 NPP units expected to make the transition in the future.

Existing NRC regulations (e.g., Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities") require licensees to design and protect systems important to safety from the effects of internal flooding. As part of its routine review of operating experience, the NRC previously recognized that seismic failures of water carrying systems in a NPP could lead to the flooding of safety-related equipment rooms and ultimately lead to failure of the equipment in those rooms. On November 7, 2005, the NRC issued Information Notice (IN) 2005-30, "Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design" (ADAMS Accession No. ML052020249). The purpose of the IN was to alert NPP licensees to the importance of establishing and maintaining the plant flooding analysis and design, consistent with NRC requirements and principles of effective risk management, to ensure that internal flooding risk is effectively managed. As part of the NRC's normal baseline inspection program, NRC inspectors semiannually select one or two plant areas and inspect internal flood protection features for risk-significant structures, systems, and components, in accordance with Inspection Procedure 71111.06, "Flood Protection Measures" (ADAMS Accession No. ML083170651). Following the issuance of the IN discussed above, the NRC's Operating Experience Branch developed an operating experience smart sample in this area to further support inspections of licensee flood protection measures. As a result of the smart sample evaluation, several site-specific issues were identified and resolved, but no generic implications were identified. Overall, existing NRC requirements and oversight programs provide the staff with assurance that NPP licensees have and maintain extensive internal flooding protection and response mechanisms that would prevent or mitigate the impact of a seismically-induced flooding event.

There is extensive information available on the causes of fires and floods resulting from seismic events, although much of that information is associated with nonnuclear effects. The main contributor to risk from a SIFF event at an NPP is for a seismic event to initiate a fire or flood that could lead to the failure of a safety-related system. There have been many studies on seismic, fire, and flooding impacts to safety-related systems, each examined independently, but

there have been relatively few on the cumulative impacts of all three types of events. The studies that have examined SIFF at NPPs have found that the risk from a SIFF is difficult to quantify because of the lack of methods and data for assessing concurrent hazards. To ensure that potential risks from seismic events are appropriately addressed, the staff is currently taking action on NTTF Recommendation 2.1, which requires sites with relatively higher seismic risk to conduct a seismic PRA. This PRA may identify systems or components that need to be upgraded to prevent detrimental effects. Given that SIFF events are a second order effect from seismic events, the staff believes that the risk of a SIFF would not be large enough to justify additional safety enhancements.

Following the accident at Fukushima Dai-ichi, the NRC took a number of regulatory actions to enhance safety at NPPs, which the staff believes would also enhance the ability to protect against a SIFF event. For example, on March 12, 2012, the staff issued a request for information pursuant to 10 CFR 50.54(f), requesting that NPP licensees perform a detailed inspection, or "walkdown," of their currently installed seismic and flooding protection features. As part of these walkdowns, NPP licensees had to ensure that these features not only met current requirements, but also identify, correct, and report any degraded conditions. The industry developed—and the NRC endorsed—guidance documents Nuclear Energy Institute 12-07, "Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features," issued May 2012, and Electric Power Research Institute 1025286, "Seismic Walkdown Guidance," issued June 2012 (ADAMS Accession No. ML12188A031), to conduct these walkdowns. During the conduct of the seismic walkdowns, NPP licensees specifically looked for equipment or systems that could have seismically-induced interactions leading to a fire or flood.

As noted in the seismic walkdown guidance, an example of a seismically-induced fire interaction could include situations where high voltage equipment has relative motion against an adjacent support structure with different foundations, which could lead to an electrical short and ultimately a fire. An example of a seismically-induced flooding interaction could include a situation where there is a long unsupported span of threaded fire protection piping, which could impact an adjacent structure and thus lead to failure of the pipe and flooding. All NPP licensees have completed these walkdowns, and the staff has conducted inspections of them using TI-2515/188, "Inspection of Near-Term Task Force Recommendation 2.3, Seismic Walkdowns," dated July 6, 2012 (ADAMS Accession No. ML12156A052), to independently verify that each licensee's seismic walkdown activities used NRC-endorsed methods. As part of these inspections, the staff did a select review of the licensee walkdowns to evaluate whether any potential SIFF vulnerabilities were identified and found that several site-specific issues were noted. All of the noted observations were entered into the licensee's corrective action program. No violations of regulatory requirements related to SIFFs were identified and NRC inspectors are following up on these corrective actions as part of the normal baseline inspection program to ensure their proper resolution in accordance with NRC requirements.

In addition to the post-Fukushima initiatives discussed above, on March 12, 2012, the staff issued Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (ADAMS Accession No. ML12054A735), which requires all NPP licensees to develop, implement, and maintain guidance and strategies to maintain key safety functions, including core cooling, containment function, and spent fuel pool cooling, following an extreme external event which causes an extended loss of all AC

power and a loss of normal access to the ultimate heat sink. The mitigation strategies being put in place in response to this order will ensure that licensees have diverse and flexible means of responding to extreme events, regardless of their origin, which will provide additional protection against detrimental effects from SIFFs. The mitigation strategies are expected to use a combination of currently installed equipment (e.g., steam-powered pumps), additional portable equipment that is stored on-site, and equipment that can be flown or trucked in from off-site support centers. Additionally, the NRC and licensees are taking action to ensure that the mitigation strategies address the reevaluated flooding and seismic hazards being developed as part of NTTF Recommendation 2.1. Overall, NRC and licensee initiatives to address lessons learned from the Fukushima Dai-ichi accident have resulted in a substantial increase in NPP safety.

The NRC maintains an operating experience program to identify issues that have occurred at NPPs and which may have some relevance to currently operating reactors. A small number of events at NPPs across the world have involved SIFFs, none of which have occurred in the United States. For example, on July 16, 2007, a strong earthquake impacted the Kashiwazaki-Kariwa NPP in Japan. Following the earthquake, a fire was initiated in a non-safety-related electrical transformer. Because of the widespread damage from the earthquake, including damage to the fire protection system, it took approximately 2 hours to extinguish the fire. There was no damage to the reactor from this event. Additionally, following the same earthquake that impacted the Fukushima Dai-ichi site on March 11, 2011, the Onagawa NPP in Japan experienced an arcing fault in a nonemergency switchgear cabinet resulting in a fire. Due to difficulties in locating and accessing the source of the fire, it took approximately 8 hours to extinguish the fire. However, as with the event at the Kashiwazaki-Kariwa NPP, there was no damage to the reactor from this event. The staff also notes that the Mineral, Virginia earthquake on August 23, 2011, which exceeded the North Anna NPP safe-shutdown earthquake, did not lead to any SIFF events nor any functional damage at that plant, as documented in “North Anna Power Station, Units 1 and 2 – Technical Evaluation of Restart Readiness Determination Plan,” dated November 11, 2011 (ADAMS Accession No. ML11308B404). In summary, the staff’s review of domestic and international operating experience did not identify any events that would support additional regulatory action in this area.

Fire and Flooding Response Procedures

As discussed above, during the ACRS review of the NTTF recommendations, documented in a letter entitled, “Initial ACRS Review of: (1) The NRC Near-Term Task Force Report on Fukushima and (2) Staff’s Recommended Actions to Be Taken Without Delay,” dated October 13, 2011 (ADAMS Accession No. ML11284A136), the ACRS recommended that NPP fire response procedures be integrated with the EOPs, SAMGs, and EDMGs. The main rationale for integrating these procedures is to better coordinate and integrate operator responses during challenging plant conditions. The ACRS subsequently expanded this recommendation to include flood response procedures in a letter dated April 22, 2015, “Draft SECY Paper, ‘Proposed Rulemaking: Mitigation of Beyond-Design-Basis Events’” (ADAMS Accession No. ML15111A271).

In SECY-15-0065, “Proposed Rulemaking: Mitigation of Beyond-Design-Basis Events (MBDBE) (RIN 3150-AJ49),” dated May 15, 2015 (ADAMS Accession No. ML15049A201), the staff discussed the potential imposition of requirements for SAMGs, including the integration with

EOPs. In the proposed rulemaking, the staff proposed a requirement for licensees to develop, implement, and maintain SAMGs, consistent with NTTF Recommendation 8. In SRM-SECY-15-0065, dated August 27, 2015 (ADAMS Accession No. ML15239A767), the Commission directed the staff to remove the proposed requirements for SAMGs and their integration with EOPs and other emergency response procedures. The key rationale for eliminating this proposed requirement is that the imposition of a requirement for SAMGs would not have resulted in a substantial safety benefit due to the low risk of events leading to severe accidents. Similarly, the staff has not identified that a substantial safety benefit would be obtained from integrating fire and flood response procedures with other response procedures. This finding is based, in part, on extensive training for operators and response personnel on the use of existing fire and flood response procedures, coupled with the benefits achieved through other fire and flood protection regulatory enhancements, such as the increasing use of risk information and NFPA 805 implementation.

The NRC also considered the following in its assessment of the recommendation to integrate fire and flood response procedures:

1. Fire response procedures are an integral part of a NPP fire protection plan. As stated in 10 CFR 50.48(a)(1), "This fire protection plan must: […] (iv) Outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage."

2. The NRC-required fire protection program is designed to function autonomously with other activities and is supported by a fire brigade that is manned in all modes of operation and is well-trained. Firefighting activities are led by personnel with knowledge of overall plant operations, including the equipment necessary for safe shutdown of the plant. These personnel communicate with the main control room to prioritize activities and ensure that they do not conflict.

3. Comprehensive firefighting strategies and implementing procedures have been developed for each area of the plant and fire brigade qualified individuals participate in drills on a quarterly basis to demonstrate proficiency with the use of these strategies and procedures in the context of concurrent use of other, nonintegrated procedures throughout the plant. Similar protocols are typically in place to respond to internal flooding events.

4. EOPs, EDMGs, FLEX Support Guidelines, and SAMGs account for equipment lost due to concurrent fires and floods (or other causes) during events by providing alternative methods to accomplish the functions the equipment was to have performed.

5. Licensee activities associated with fire and flood protection and response are subject to NRC oversight as part of the NRC's inspection program.

<u>Summary of Staff's Assessment</u>

As discussed above, the staff has determined that regulatory requirements for enhancements to the existing capabilities to prevent or mitigate SIFFs could not be justified. Given the broad regulatory activities pertaining to seismic, fire, and flooding events, operating experience

involving SIFFs, and actions taken in response to the Fukushima accident, the staff's conclusion is that additional requirements are not needed to ensure continued safe operation of U.S. NPPs.

**Stakeholder Interactions**

As discussed above, the staff has interacted extensively with both internal and external stakeholders on the Tier 1 component of this recommendation (development of a PRA methodology).

The NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015.

**Conclusion**

Subject to Commission approval and based on the staff's assessment provided above, the staff is closing Recommendation 3. The staff intends to continue work to finalize the PRA feasibility study, with a goal of completing that study by the end of Calendar Year 2015. The staff will inform the Commission if any insights obtained as part of the completion of that study change the results of its assessment of Recommendation 3.

**Enclosure 4: Proposed Resolution Plan for Tier 3 Recommendations 5.2 for Reliable Hardened Vents Other Containments and 6.0 for Hydrogen Control and Mitigation Inside Containment and Other Buildings**

**Background**

As described in SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated December 23 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11186A950), the U.S. Nuclear Regulatory Commission's (NRC's) Near-Term Task Force (NTTF) identified Recommendation 5.2, which recommended that the NRC assess the need to require the installation of reliable, hardened venting systems for containments with designs other than Mark I and II (which are addressed as part of Recommendation 5.1). The NTTF also recommended that the staff assess the need to further strengthen requirements associated with hydrogen control and mitigation inside and outside reactor containment buildings as part of NTTF Recommendation 6. In SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011 (ADAMS Accession No. ML11272A111), the staff prioritized these as Tier 3 activities, since they required further staff study and the insights from implementation of Recommendation 5.1 and related international activities to support a regulatory decision.

In SECY-11-0137, the NRC staff described its proposals for immediate regulatory actions and longer-term evaluations to address the NTTF recommendations. Among the highest-priority Tier 1 actions that the NRC staff proposed was the issuance of orders to address Recommendation 5.1, requiring reliable hardened containment vents for those licensees of boiling water reactors (BWRs) with Mark I and II containment designs. Venting Mark I and II containments can help prevent the loss of, and facilitate recovery of, important safety functions, such as reactor core cooling, reactor coolant inventory control, containment cooling, and containment pressure control. The NRC issued Order EA-12-050 on March 12, 2012 (ADAMS Accession No. ML12054A694), requiring reliable, hardened vents for these plants. The NRC subsequently revised these requirements by Order EA-13-109, dated June 6, 2013 (ADAMS Accession No. ML13130A067), to make the venting systems for Mark I and II containments capable of operation during severe accident conditions.

The NRC staff has been actively participating in various international studies, including a working group studying hydrogen generation, transport, and risk management organized by the Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA). The NRC staff has also gathered insights from other Fukushima-related activities, as well as probabilistic risk studies, previous evaluations of generic issues, operating experience, and other available information. These insights are being used to help assess whether additional studies of containment performance and the control of hydrogen following potential severe reactor accidents would justify regulatory actions beyond those already taken for plants with Mark I and II containments.

Containment performance and the control of hydrogen have been the focus of a number of previous NRC studies and evaluations. In addition to the recent evaluations related specifically to Mark I and II containments, the NRC completed detailed assessments as part of the Containment Performance Improvement Program (CPIP) in the 1980s, resolved generic safety

issues, and established requirements such as Section 50.44, “Combustible Cas Control for Nuclear Power Plants,” in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of production and Utilization facilities.” Containment performance and hydrogen-related issues have also been addressed in major studies, such as those documented in NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” issued December 1990; and NUREG-1935, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Report,” issued November 2012. The NRC staff described the CPIP effort in SECY-88-147, “Integration Plan for Closure of Severe Accident Issues,” dated May 25, 1988. This effort evaluated generic severe accident challenges for each light water reactor (LWR) containment type to determine whether additional regulatory guidance or requirements concerning containment features were warranted. Therefore, the CPIP is especially relevant to this evaluation. The CPIP was initiated to address uncertainties in the ability of LWR containments to successfully survive some severe accident challenges, as indicated by the results documented in NUREG-1150. All LWR containment types were assessed in the CPIP, but as in more recent evaluations, many of the activities were focused on Mark I and II containment designs. The CPIP identified potential improvements for Mark I and II designs, and some of these features were further enhanced through the Tier 1 activities associated with Orders EA-12-049, “Mitigating Strategies for Beyond Design Basis External Events,” dated August 29, 2012 (ADAMS Accession No. ML12054A735); and EA-13-109, “Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions for BWR Mark I and Mark II Containments,” dated June 6, 2013 (ADAMS Accession No. ML13130A067). As described in NUREG-0933, “Resolution of Generic Safety Issues,” published December 2011, the NRC staff did not identify generic improvements that would apply to Mark III, ice condenser, or large dry containments. Rather, the staff requested that licensees with plants with these containment designs consider insights from the CPIP within the individual plant evaluations.

The NRC has also addressed containment performance issues and the role of the containment in limiting the consequences of severe accidents in research programs, resolving generic safety issues, and evaluating regulatory actions that were ultimately not pursued because the possible action was found to provide only minimal safety benefits. Many of the NRC-sponsored research projects related to containment performance are described in NUREG/CR-6906, “Containment Integrity Research at Sandia National Laboratories,” published in July 2006. NUREG-0933 describes the NRC’s assessment and closure of various containment-related issues, including the activities within the CPIP. Containment performance has also been the focus of regulatory requirements within plant technical specifications and in the development of regulations such as 10 CFR 50.44. The NRC staff evaluated various issues and potential improvements to containment performance as part of internal initiatives (e.g., SOARCA) and in response to petitions for enforcement action or rulemaking. These activities have collectively added to the body of knowledge related to containment performance and the control of hydrogen following severe reactor accidents. The activities undertaken in response to the Fukushima accident provide additional insights and have resulted in regulatory actions, such as issuance of Orders EA-12-049 and EA-13-109, which further enhance safety.

**Current Status**

The staff has performed a preliminary analysis to determine if there is a need for regulatory action associated with Recommendations 5.2 and 6. The initial conclusion of this analysis is that further regulatory action or study beyond those completed for Mark I and Mark II

containments is not warranted. Details of this analysis are provided in this enclosure, along with the staff's plans to obtain stakeholder input, finalize its analysis, and complete its evaluation of these recommendations. In the absence of new information from external stakeholders, the staff expects these additional activities will support and provide further justification for the initial conclusion.

BWR Mark I and II Containments

In March 19, 2013, staff requirements memorandum (SRM) (ADAMS Accession No. ML13078A017) to SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling-Water Reactors with Mark I and Mark II Containments," dated November 26, 2012, the Commission directed the NRC staff to: (1) issue a modification to Order EA-12-050 requiring BWR licensees with Mark I and II containments to upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions, and (2) develop a technical basis and rulemaking for filtering strategies with drywell filtration and severe accident management for BWR Mark I and II containments. The staff subsequently issued Order EA-13-109[1], which rescinded the requirements imposed by Order EA-12-050 and replaced them with the following requirements for licensees with BWRs with Mark I and II containments:

- Phase 1: Upgrade the venting capabilities from the containment wetwell to provide reliable, severe accident capable hardened vents to assist in preventing core damage and, if necessary, to provide venting capability under severe accident conditions.

- Phase 2: Install a reliable, severe accident capable drywell vent, or develop a reliable containment venting strategy that makes it unlikely the site would need to vent from the containment drywell during a severe accident.

The NRC's interim staff guidance (ISG) for Phase 1 of the order was issued in November 2013, which endorsed the guidance developed by the Nuclear Energy Institute (NEI) and an industry working group, NEI 13-02, Revision 0 (ADAMS Accession No. ML13316A853). The NRC issued the ISG for Phase 2 requirements in April 2015. This ISG endorsed the updated industry guidance document, NEI 13-02, Revision 1 (ADAMS Accession No. ML15113B318). As required by Order EA-13-109, licensees with Mark I and II containments submitted their overall integrated plans (OIPs) for Phase 1 by June 30, 2014. The staff has completed its review of Phase 1 plans and has issued interim staff evaluations. Licensees are required to submit OIPs for Phase 2 of EA-13-109 by December 31, 2015.

Containment Protection and Release Reduction (CPRR) Rulemaking

As directed by the SRM for SECY-12-0157, the staff assessed possible additional requirements

[1] Order EA-13-109 states that the requirement to provide a reliable hardened containment vent system (HCVS) to prevent or limit core damage upon loss of heat removal capability is necessary to ensure reasonable assurance of adequate protection of public health and safety (maintaining the justification in Order EA-12-050), while the requirement that the reliable HCVS remain functional during severe accident conditions is a cost-justified substantial safety improvement under 10 CFR 50.109(a)(3).

for containment pressure control and venting, to include measures to enhance the capability to maintain containment integrity and to cool core debris. These evaluations formed the draft regulatory basis prepared for the CPRR rulemaking. The main objective of the CPRR regulatory basis was to determine what, if any, additional requirements are warranted related to filtering strategies and severe accident management of BWRs with Mark I and II containments assuming the installation of severe accident capable hardened vents per Order EA-13-109. The staff interacted with external stakeholders and identified four major alternatives for possible courses of action related to filtering strategies and severe accident management for BWRs with Mark I and II containment designs. The CPRR alternatives were the following:

1. Take no action (Order EA-13-109 implemented without additional regulatory actions related to Mark I and II containments).

2. Pursue rulemaking to make Order EA-13-109 generically applicable to protect Mark I and II containments against overpressurization.

3. Pursue rulemaking to address overall Mark I and II containment protection against multiple failure modes by making Order EA-13-109 generically applicable and requiring external water addition points that would allow for water addition into the reactor pressure vessel (RPV) or drywell (DW) to prevent containment failure from both over-pressurization and liner melt-through.

4. Pursue rulemaking to address both containment protection against multiple failure modes and release reduction measures for controlling releases through the containment venting systems. This alternative would include making Order EA-13-109 generically applicable and require external water addition into the RPV or DW. In addition, licensees would be required to reduce the fission products released from the containment by either implementing strategies to maximize the availability and efficiency of the wetwell in scrubbing or filtering fission products before venting from containment and/or installing an engineered filter in the containment vent paths.

The draft regulatory basis document was provided to the Commission in SECY-15-0085, “Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities,” dated June 25, 2015 (ADAMS Accession No. ML15022A218). In the SRM to SECY-15-0085, dated August 19, 2015 (ADAMS Accession No. ML15231A471), the Commission directed the staff to take no further action beyond those associated with implementation of Order EA-13-109 (Alternative 1) because the addition of engineered filters would not provide a substantial additional safety benefit and the safety benefits of severe accident water addition are being provided by licensees for compliance with the Order. In addition, the SRM directed the staff to leverage the draft regulatory basis to the extent applicable to support resolution of the post-Fukushima Tier 3 item related to containments of other designs (i.e., Recommendation 5.2).

<u>International Activities</u>

The NRC staff has participated in various international meetings and working groups related to reactor containment performance and has used insights from these activities to identify and evaluate technical and regulatory issues. For example, in “Staff Requirements – Briefing of the

Status of Lessons Learned from the Fukushima Dai-ichi Accident," dated August 24, 2012 (ADAMS Accession No. ML122400033), the Commission directed the staff to compare practices for hydrogen control for plants in other countries with those of U.S. plants. The staff from the NRC Office of Nuclear Regulatory Research participated as members of an OEDC/NEA working group conducting a study of hydrogen generation, transport, and risk management. The working group issued a report entitled, "Status Report on Hydrogen Management and Related Computer Codes," in June 2014. The report describes various containment designs, national requirements, and actions addressing lessons learned from the Fukushima accident. Measures to control hydrogen during severe accidents, including the use of passive autocatalytic recombiners, have been taken or are being pursued for many foreign plants. Currently, some countries are assessing the need for hydrogen mitigation measures outside containment, but no specific requirements have been imposed in most countries for such measures. The OECD/NEA report provides a comparison of various designs and practices for plants in the United States and other countries.

**Discussion**

The staff has used the insights from the technical evaluations discussed above in developing its initial assessment of Recommendations 5.2 and 6. The staff has also considered previous Commission decisions on post-Fukushima matters, regulatory analysis, and longstanding policies related to safety goals and treatment of severe accidents for operating reactors. These decisions are provided in SRMs related to a number of papers, such as the following:

- SECY-12-0110, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," dated August 21, 2012 (ADAMS Accession No. ML12173A478)

- SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated November 26, 2012

- COMSECY-13-0030, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated November 25, 2013 (ADAMS Accession No. ML13329A918)

- SECY-15-0065, "Mitigation of Beyond-Design-Basis Events," dated May 15, 2015 (ADAMS Accession No. ML15049A201)

- SECY-15-0085, "Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities," dated June 18, 2015 (ADAMS Accession No. ML15022A218)

These decisions provide continuity between current assessments and previous evaluations of containment-related safety issues and maintain the relevance of previous regulatory and backfit analyses and the associated decisions.

Table 1 provides a summary of key functional areas related to these two recommendations and how they have been addressed for each containment type, with a more detailed assessment for each containment type provided below.

Table 1. Recommendation 5.2 and 6 – Other Containment Designs and Hydrogen Control Requirements and Practices

| | Core Cooling Functions | Venting/Heat Removal for Containment Pressure Control | | Other Containment Failure Modes/Core Debris Cooling | Release Reduction (Filtering) | Hydrogen Control | |
|---|---|---|---|---|---|---|---|
| | | Pre-Core Damage | Severe Accident | | | Containment | Other |
| Mark I | EA-12-049<br>EA-13-109 | EA-13-109<br>EA-12-049<br>EOPs, FSGs | EA-13-109<br>SAMGs | EA-13-109<br>(CPRR) | N/A<br>(CPRR) | EA-13-109<br>SAMGs | EA-13-109<br>SAMGs |
| Mark II | EA-12-049<br>EA-13-109 | EA-13-109<br>EA-12-049<br>EOPs, FSGs | EA-13-109<br>SAMGs | EA-13-109<br>(CPRR) | N/A<br>(CPRR) | EA-13-109<br>SAMGs | EA-13-109<br>SAMGs |
| Mark III | EA-12-049 | EA-12-049<br>EOPs, FSGs | SAMGs | SAMGs | N/A<br>(current assessment) | GSI-189<br>EA-12-049<br>SAMGs, FSGs | GSI-189<br>EA-12-049<br>SAMGs,<br>FSGs |
| Ice Condenser | n/a | EOPs | SAMGs | SAMGs | N/A<br>(current assessment) | GSI-189<br>EA-12-049<br>SAMGs, FSGs | GSI-189<br>EA-12-049<br>SAMGs,<br>FSGs |
| Large Dry | n/a | EOPs | SAMGs | SAMGs | N/A<br>(current assessment) | SAMGs | N/A<br>(current assessment) |

EA-12-049: Mitigation Strategies Order
EOPs: Emergency Operating Procedures
SAMGs: Severe accident management guidelines
EA-13-109: BWR Mark I/II Severe accident capable vent order
FSGs: FLEX (Mitigating Strategies) Support Guidelines
GSI-189: Generic Safety Issue re: Hydrogen Issues

## **BWR Mark I and II**

### Containment Performance

As part of implementing Order EA-13-109 requirements, licensees are planning to install a wetwell venting system that remains functional under severe accident conditions, and an approach involving severe accident water addition (SAWA). Licensees are expected to implement a severe accident water management (SAWM) strategy to control the water levels in the suppression pool, so that it is unlikely a licensee would need to vent from the containment drywell during severe accident conditions. The NRC staff and industry evaluations have shown that the SAWA/SAWM strategies not only serve the purpose of Order EA-13-109 requirements, but could also prevent containment failure from mechanisms other than overpressurization. As part of developing the CPRR regulatory basis, the staff analyzed numerous alternatives in relation to quantitative health objectives (QHOs) described in the Safety Goal Policy Statement. As shown in Figure 1, taken from the CPRR draft regulatory basis document, significant margins

are calculated between existing plant risks from an extended loss of electrical power and the NRC's safety goals; therefore, changes to Mark I and II containments beyond those required by Order EA-13-109 would not constitute substantial safety improvements.

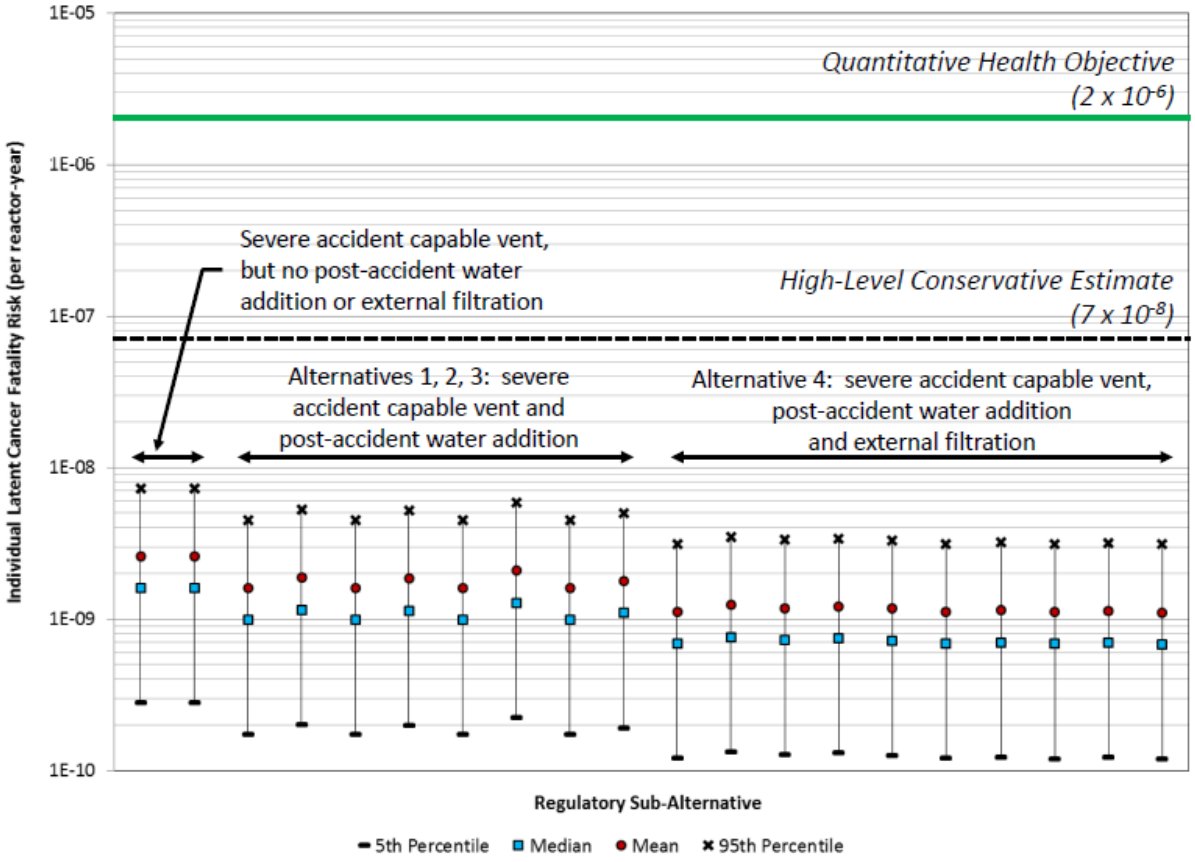



Figure 1

The Commission's SRM related to SECY-15-0085 directed the staff to discontinue further CPRR-related rulemaking activities. The actions taken under Order EA-13-109 and decisions made on SECY-15-0085 have resolved the shaded areas in Table 1 for Mark I and Mark II containments.

Hydrogen Control

The issue of hydrogen control in Mark I and II primary containments was considered within the technical analyses supporting the severe accident functions of Order EA-13-109 and the consideration of additional containment performance issues as part of the CPRR regulatory basis document. Mark I and II containments are inerted during normal operations to address the requirements of 10 CFR 50.44. Risk evaluations and insights from the Fukushima accident show that some severe accident sequences can generate large amounts of hydrogen and explosions can occur if measures are not taken to control or prevent them. The Fukushima

accident also highlighted the possible migration of hydrogen to buildings outside the primary containment and the need to evaluate possible features or procedures to prevent explosions in the reactor building or other structures.

The NRC staff performed detailed evaluations of possible severe accident conditions, including the generation of hydrogen and other combustible gases within Mark I and II containments, as part of the work supporting Order EA-13-109 and the CPRR draft regulatory basis document. Similar studies performed by the industry are documented in the Electric Power Research Institute report, "Technical Basis for Severe Accident Mitigating Strategies, Volume 1," issued April 2015. In the CPRR draft regulatory basis document, the staff described the benefits of improved venting operations for the control of hydrogen in the primary containment and other buildings as follows:

> The behavior of hydrogen in the containment is shown in Figure 4-19 [Figure 2 below], "Mark I Hydrogen Generation and Transport for Case 9 (SAWA)." The blue line represents the total hydrogen generation which should be almost identical with the amount remaining inside the containment and the amount that is vented (represented by the green line). The amount of hydrogen that remains inside the containment (both the drywell and the wetwell air space as shown by the red line) quickly decreases as a result of venting. With the wetwell vent open during the transient, the total amount of hydrogen is kept very low in the long term (below 30 kg). Therefore, containment venting is very efficient in purging the hydrogen from the containment. The presence of water seems to avoid containment failure and any uncontrolled release of hydrogen to the reactor building which remains intact for the duration of the accident.

**Figure 4-19: Mark I Hydrogen Generation and Transport for Case 9 (SAWA)**

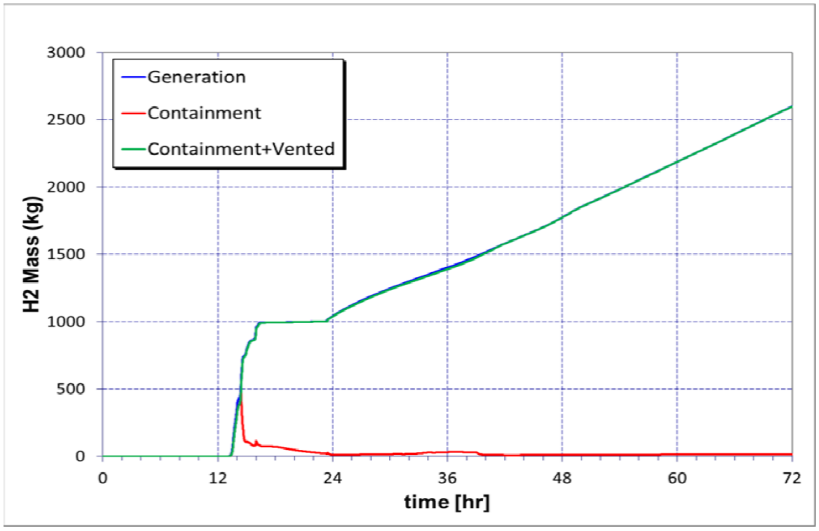



Figure 2

The technical analyses for Order EA-13-109 and the CPRR draft regulatory basis show that the threat of explosions from combustible gases is significantly reduced by effective venting strategies and the SAWA/SAWM approaches being taken as part of implementing the order.

Severe accident management guidelines (SAMGs) maintained by licensees and updated after the Fukushima accident also include specific measures to monitor and vent Mark I and II containments to address hydrogen issues. The enhancements provide some further risk reductions by improving the control of hydrogen in Mark I and II containments, even though more specific regulatory actions would likely not be justified given the large margins between plant risks and the NRC's safety goals (as shown in Figure 1). Further evaluations to identify other possible improvements for hydrogen control in primary containments or other buildings would likewise be unlikely to justify the imposition of additional regulatory requirements. As such, the staff's initial assessment is that Recommendation 6 can be closed for Mark I and II containments.

**BWR Mark III**

Containment Performance

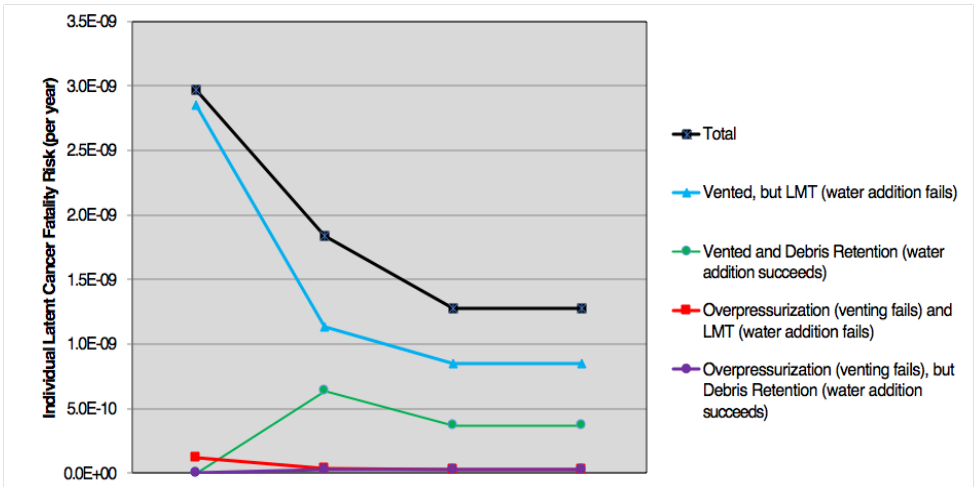
There are four operating BWRs with Mark III containments located at four sites in the United States. The Mark III containment is approximately five times the volume of the Mark I containment and 65 to 85 percent of the volume of a large dry pressurized water reactor (PWR) containment. The containment design pressure of a Mark III containment is 15 pounds per square inch gauge (psig) (25 percent of a Mark I and 30 percent of a large dry containment). Unlike Mark I and II containments, the Mark III containment is not inerted, but instead has igniters for hydrogen control. The NRC evaluated a Mark III containment (Grand Gulf) as part of the activities associated with NUREG-1150. Supporting evaluations of containment issues for Mark III containments are described in NUREG/CR-5529, "An Assessment of BWR Mark III Containment Challenges, Failure Modes, and Potential Improvements in Performance," published in January 1991. The modern BWR design incorporating the Mark III containments includes a diversity of ways to provide water to the core, and therefore, reactors with this type of containment have a relatively low estimated core damage frequency (on the order of $10^{-6}$/year). The pre-Fukushima evaluations of core damage and containment performance for licensed Mark III plants did not identify generic improvements that warranted regulatory actions (see NUREG-0933).

Mark III containments are pressure suppression containments and have system interactions between the core cooling and containment functions, similar to plants with Mark I and II containments. These interactions are especially important during an extended loss of electrical power when cooling systems used for design-basis accidents are not available. Order EA-12-049 requires all operating plants to develop mitigating strategies for events involving extended losses of electrical power and loss of normal access to the plant's ultimate heat sink. The mitigating strategies include three phases: (1) an initial phase which must be survived with installed equipment such as steam-driven pumps; (2) a transition phase which uses portable, onsite equipment; and (3) a final phase which may credit offsite resources.

Suppression pool cooling is an important safety function within the mitigating strategies for the plants with Mark III containments. Venting is not a primary method for suppression pool cooling for three of the Mark III plants, while one does include venting from the suppression pool as part of its mitigating strategies. Instead, most plants with Mark III containments have included in their mitigating strategies additional capabilities to power suppression pool cooling equipment (i.e., through the use of portable power supplies). The NRC staff has reviewed these

approaches and prepared interim staff evaluations documenting that the licensees for Mark III plants have developed an acceptable approach for addressing core cooling functions and containment pressure control, including the need to remove heat from the suppression pool. These requirements address the functions deemed necessary to ensure reasonable assurance of adequate protection of public health and safety in Order EA-13-109 issued to plants with Mark I and Mark II containments. The estimated low frequency for extended losses of electrical power makes it unlikely that further evaluations of means to cool or vent suppression pools in Mark III containments beyond those required under Order EA-12-049 would identify a cost-justified substantial safety improvement.

The activities supporting implementation of Order EA-13-109 for Mark I and II containments highlighted the need to take a holistic approach to considering improvements to containment performance during potential severe accidents. An important insight from the CPRR activities is that potential safety benefits from improvements to address some failure mechanisms or reduce releases by adding engineered filters can be limited by other potential failure mechanisms and accident sequences. This is demonstrated in Figure 3 for Mark I and II containments, which shows that the benefits from engineered filters are limited by factors such as the possible failure of equipment leading to releases that would not be scrubbed effectively by the filters. Similar relationships and limitations would likely apply to Mark III reactor designs and thereby limit the potential effectiveness of specific items related to improving containment performance during severe accidents. NUREG/CR-5529, “Reevaluation of Station Blackout Risk at Nuclear Power Plants,” issued December 2005, evaluated potential severe accident improvements for Mark III containments and informed the NRC’s decision that no regulatory actions were warranted, except additional consideration for improving the control of hydrogen (see next section). Insights from the Fukushima accident do not undermine the findings from these previous evaluations.




| Sub-Alternative | 1 | 3A | 4Bi(1) | 4Ci(1) |
|---|---|---|---|---|
| water addition capability | no | yes | yes | yes |
| filter | no | no | DF=10 | DF=1000 |
| CCFP | 100% | 42% | 42% | 42% |

CCFP = conditional containment failure probability (overpressurization and/or LMT)
LMT = liner melt-through

Figure 3

In summary, the NRC staff concludes that additional detailed study of possible improvements to the performance of Mark III containments during the mitigation of events or during severe accident conditions would be unlikely to identify regulatory actions meeting the threshold for a substantial safety improvement. Therefore, the staff's initial assessment is that Recommendation 5.2 can be closed for Mark III containments with no additional requirements beyond those imposed by Order EA-12-049. In the absence of new information from external stakeholders, the staff expects that plans for additional evaluations of severe accident scenarios and associated risks relative to the QHOs will support and provide further justification for the initial conclusion.

Hydrogen Control

NUREG-1150 and other studies identified hydrogen issues as a potential concern for Mark III containments. The evaluations documented in NUREG/CR-6427, "Assessment of the DCH [direct containment heating] Issue for Plants with Ice Condenser Containments," issued April 2000, led to GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion during a Severe Accident," updated February 4, 2009, and assessments of potential safety enhancements related to the reliability of ignitor systems. To deal with large quantities of hydrogen, ice condenser and Mark III containments are equipped with alternating current (ac) powered igniters, which are intended to control hydrogen concentrations in the containment atmosphere by initiating limited "burns" of hydrogen. In essence, the igniters prevent the hydrogen (or any other combustible gas) from accumulating in large quantities and then suddenly burning (or detonating) all at once, which could pose a threat to containment integrity. For most accident sequences, the hydrogen igniters can deal with the potential threat from combustible gas buildup. The situation of interest for GSI-189 related primarily to accident sequences associated with station blackouts, where the igniter systems are not available because they are ac-powered. Thus, the concern does not affect the frequency of severe accidents, but does affect the likelihood of a significant release of radioactive material to the environment should such an accident occur.

Because this issue was not incorporated into the original scope of security-related modifications implemented following the September 11, 2001, terrorist attacks, the staff held meetings with licensees to further explore the proper consideration of security insights in providing backup power to the ignitor systems. The staff reviewed industry proposals and concluded that the proposed modifications would resolve GSI-189 and provide benefit for some security scenarios. On April 23, 2007, the NRC's Executive Director for Operations issued a memorandum informing the Commission of the staff's intent to accept the commitments associated with providing backup power to hydrogen ignitors and perform verification inspections at the affected sites. On June 15, 2007, the NRC staff issued letters to affected licensees accepting these commitments. The regulatory commitments related to backup power to the ignitor systems received additional attention during the development of guidance for Order EA-12-049. The guidance documents for compliance with Order EA-12-049 identify the backup power supplies to the ignitor systems for ice condenser and Mark III containments to be part of the containment protection features within the scope of the order. By improving the reliability of ignitor systems during station blackout scenarios, the actions taken provide confidence that combustible gases

will not cause a loss of primary containment integrity and reduce the chances that they will migrate to other structures, as occurred during the Fukushima accident.

Based on the assessments discussed above, the staff's initial evaluation concludes that the additional study of possible improvements to hydrogen control for Mark III containments or nearby structures would be unlikely to identify regulatory actions meeting the threshold for a substantial safety improvement.

**PWR Ice Condenser**

Containment Performance

There are nine operating PWRs with ice condenser containments located at five sites in the United States and an additional unit (Watts Bar Unit 2) is expected to enter commercial operation in the future. The volumes and design pressures for ice condenser containments are similar to the Mark III BWR containments. Ice condenser containments also have igniters for hydrogen control. The NRC evaluated an ice condenser containment (Sequoyah) as part of the activities associated with NUREG-1150. Supporting evaluations of containment issues for ice condenser containments are described in various studies, including NUREG/CR-6427. The pre-Fukushima evaluations of core damage and containment performance for licensed ice condenser plants did not identify generic improvements that warranted regulatory actions (see NUREG-0933).

Ice condenser containments are pressure suppression containments, but like other PWR designs, they do not have direct system interactions between core cooling functions and containment functions, as discussed above for BWRs. As discussed above, Order EA-12-049 requires all operating plants to develop a three phase approach for mitigating events involving extended losses of electrical power and loss of normal access to the ultimate heat sink. Venting is not a primary method for protecting ice condenser containments as part of compliance with Order EA-12-049. Rather, these plants use the presence of the ice and containment sprays to maintain containment pressure and temperature within limits during an extended loss of electrical power. The NRC staff has reviewed these approaches and prepared interim staff evaluations documenting that the licensees with ice condenser plants have developed an acceptable approach for addressing core cooling and containment functions. The estimated low frequency for extended losses of electrical power and expected plant response to the loss of power scenarios, including the implementation of mitigating strategies, makes it unlikely that further evaluations of the means to protect ice condenser containments, beyond those developed for compliance with Order EA-12-049, would identify a cost-justified substantial safety improvement. Therefore, the staff's initial assessment is that Recommendation 5.2 can be closed for ice condenser containments.

Studies documented in reports such as NUREG-1150 and NUREG/CR-6427 evaluated potential severe accident scenarios for ice condenser containments and contributed to the NRC's decision that no regulatory actions were warranted, except additional consideration for improving the control of hydrogen (see next section). Insights from the Fukushima accident do not undermine the findings from these previous evaluations. The NRC staff is evaluating an ice condenser plant (Sequoyah) as part of a continuation of the SOARCA project as directed by the Commission in the SRM to SECY-12-0092 (ADAMS Accession No. ML12341A349). Should the

NRC staff continue the evaluation of this Tier 3 activity into 2016, preliminary results from the ongoing study could be used to provide further confirmation that no additional regulatory actions are warranted for ice condenser containments. In any case, significant findings from the SOARCA assessments will be shared with licensees and other stakeholders and any adverse findings would be evaluated for possible inclusion in the NRC's generic issue program.

Hydrogen Control

The evaluation of hydrogen control for ice condenser containments follows the above discussion for Mark III containments. By improving the reliability of ignitor systems during station blackout scenarios, the actions taken provide confidence that combustible gases will not cause a loss of primary containment integrity and will reduce the chances that they will migrate to other structures, as occurred during the Fukushima accident. Based on the assessments discussed above, the staff concludes that an additional study of possible improvements to hydrogen control for ice condenser containments or nearby structures would be unlikely to identify regulatory actions beyond those already taken that would meet the threshold for a substantial safety improvement.

**PWR Large Dry**

Containment Performance

There were 56 operating PWRs with large dry containments located at 33 sites in the United States. Four PWRs with AP1000 designs are under construction and are discussed in a following section for reactors licensed under the provisions of 10 CFR Part 52, Licenses, Certification, and Approvals for Nuclear Power Plants." For the sake of this discussion, large dry containments also include those maintained at sub atmospheric conditions during normal operations. A large dry containment is designed to contain the blowdown mass and energy from a large break loss of coolant accident, assuming any single active failure in the containment heat removal systems. These systems may include containment sprays and/or fan coolers, depending on the particular design. Large dry containments can be of either concrete or steel construction. All United States concrete containments have steel liners to assure leak tightness. Large dry (and all other) containments have a large, thick basemat that provides seismic capability, supports the structures, and may serve to contain molten material during a severe accident. PWR designs with large dry containments do not have direct system interactions between core cooling functions and containment functions as discussed for BWRs.

As discussed above, Order EA-12-049 requires all operating plants to develop a three phase approach for mitigating events involving an extended loss of electrical power and loss of normal access to the ultimate heat sink. Venting is not a primary method for protecting large dry containments as part of compliance with this order. Instead, plants use containment sprays or restore containment cooling functions to maintain containment pressure and temperature within limits during an extended loss of electrical power. The NRC staff has reviewed licensees' approaches for compliance with this order and prepared interim staff evaluations documenting that the licensees for plants with large dry containments have developed acceptable approaches for addressing core cooling and containment functions. The estimated low frequency for extended losses of electrical power, and the expected plant response to the loss of power scenarios, including the implementation of mitigating strategies, makes it unlikely that

further evaluations of the means to protect large dry containments beyond the overall integrated plans developed for Order EA-12-049 would identify a cost-justified substantial safety improvement. Therefore, the staff's initial assessment is that Recommendation 5.2 can be closed for large dry containments.

Large dry containments have been evaluated in terms of severe accident behavior in several major NRC studies, including NUREG-1150 and most recently in NUREG-1935 (SOARCA). Results from the SOARCA study for the PWR large dry containment pilot plant, Surry, are provided on the right side of Figure 4 in terms of individual latent cancer fatality risk. Similar to the evaluations presented in the CPRR regulatory basis document, the figure shows a significant (orders of magnitude) margin between the risks associated with an extended loss of ac power event and the QHOs defined by the NRC's Safety Goal Policy Statement. A preliminary assessment performed by the NRC staff determined that site-to-site variations related to reevaluated external hazards would not challenge the conclusion that a generic requirement is not warranted for severe accident measures beyond those already in place for large dry containments. Insights from the Fukushima accident do not undermine the findings from these previous evaluations. As directed by the Commission in the SRM to SECY-12-0092, an uncertainty analysis of the SOARCA Surry unmitigated short-term station blackout (STSBO) scenario is underway which will provide additional insights.

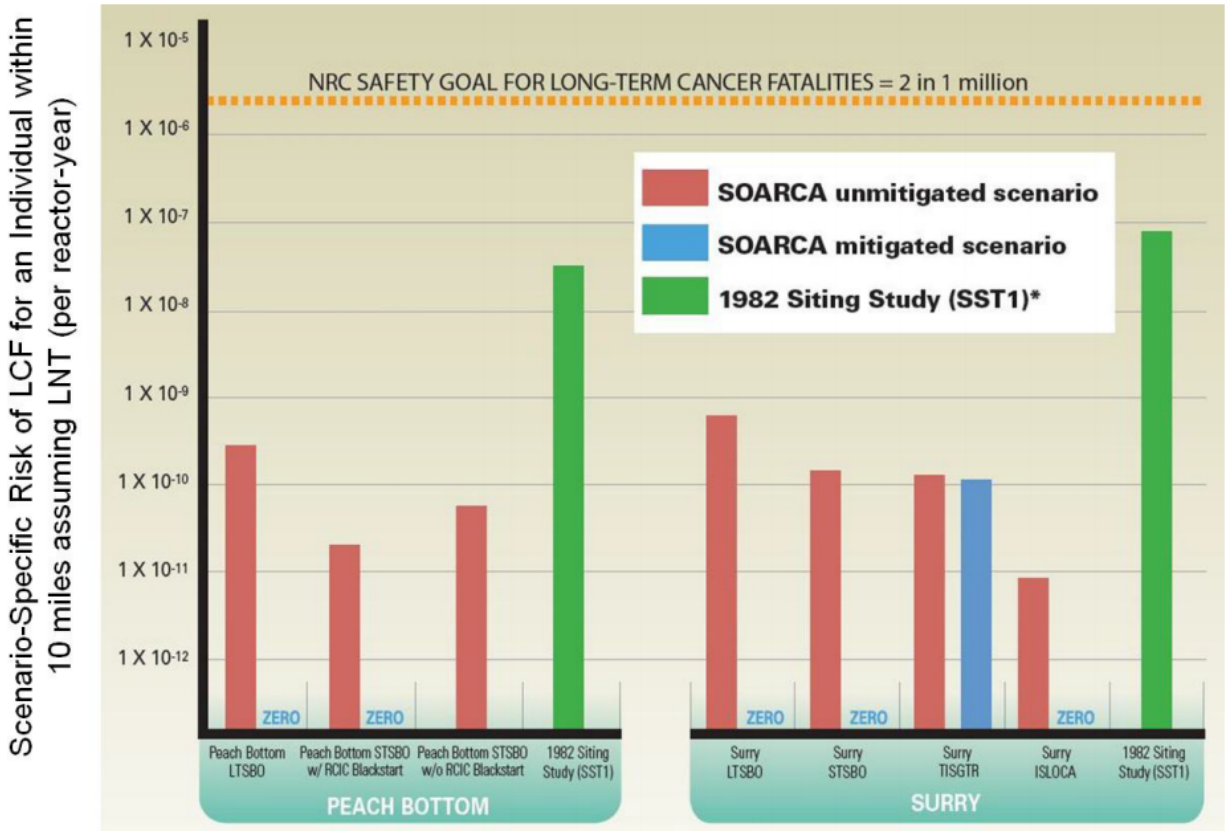


Figure 4 – Comparison of Individual Latent Cancer Fatality (LCF) Risk Results for SOARCA Scenarios to the NRC Safety Goal and to Extrapolations of the 1982 Siting Study SST1 (taken from NUREG-1935 Figure ES-3)[2]

Hydrogen Control

Evaluations of hydrogen control for large dry containments have been addressed in several NRC risk studies and documentation related to the development of 10 CFR 50.44 (SECY-03-0127), including the associated regulatory analysis. A detailed assessment is documented in NUREG/CR-5662, "Hydrogen combustion, control, and value-impact analysis for PWR dry containments," issued in June 1991. Based on the assessments discussed above, the staff concludes that additional study of possible improvements to hydrogen control for large dry containments or nearby structures would be unlikely to identify regulatory actions meeting the threshold for a substantial safety improvement.

**Reactors Licensed Under 10 CFR Part 52**

For nuclear power plants licensed under 10 CFR Part 52, including the AP1000 plants currently under construction, the NRC imposes additional requirements for containments beyond those for currently operating plants. This practice is consistent with the NRC's Severe Accident Policy Statement that new nuclear power plants should incorporate improvements during the design and construction that were not practical or cost-effective to require as modifications to existing plants. New reactors licensed under 10 CFR Part 52 must address similar design basis accidents as operating plants, but must also have severe accident design features to increase the ability of containments to maintain their integrity during severe accident conditions. In addition, more conservative hydrogen generation rates and related controls are imposed in 10 CFR 50.44 for plants licensed after 2003. The NRC staff assessed potential enhancements beyond those already included for new plants licensed under 10 CFR Part 52 and found that such measures would not likely be justified under the finality provisions established under 10 CFR Part 52 (similar to backfit requirements defined in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities").

**Stakeholder Interactions**

The NRC staff held numerous public meetings with nuclear industry representatives related to the activities for Mark I and II containments. The staff also made presentations to subcommittees and the full committee of the Advisory Committee on Reactor Safeguards (ACRS). The NRC staff interacted with other interested stakeholders during discussions on petitions for enforcement actions, at public meetings, and in correspondence related to Mark I and II containments and various proposals for improvements, including the installation of engineered filters.

The staff has had fewer public interactions specifically related to Recommendations 5.2 and 6. However, during a meeting held on October 6, 2015, the NRC staff provided the Fukushima

[2] LTSB: long-term station blackout; LNT: linear no-threshold; RCIC: reactor core isolation cooling; ISLOCA: interfacing systems loss-of-coolant accident; SST: siting source term

subcommittee of the ACRS with an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations. A similar meeting is planned with the ACRS full committee on November 5, 2015. The staff expects to conduct additional focused meetings on these recommendations with the ACRS, industry, and/or other stakeholders to support the completion of its final analysis.

**Conclusion**

Based on the evaluations described above, the staff does not expect that regulatory actions beyond those already taken is needed to close Recommendations 5.2 and 6. However, the staff plans to more fully document its basis for closing these recommendations, interact with the ACRS and external stakeholders, and provide more detailed documentation, incorporating insights from these interactions, to the Commission by March 2016.

Related to the staff's consideration of Recommendation 6, the Natural Resources Defense Council submitted a petition for rulemaking (PRM) on October 14, 2011, requesting the NRC revise 10 CFR 50.44 regarding the measurement and control of combustible gas generation and dispersal within a power reactor system (PRM-50-103). The petition addresses several issues beyond those identified in the NTTF report and will be addressed in a separate paper to the Commission.

**Enclosure 5: Proposed Resolution Plan for Tier 3 Additional Recommendation - Enhanced Reactor and Containment Instrumentation for Beyond-Design-Basis Conditions**

**Background**

As directed by staff requirements memorandum (SRM) to SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated August 19, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112310021), the staff sought to identify additional recommendations related to lessons learned from the Fukushima Dai-ichi event, beyond those identified in the Near-Term Task Force (NTTF) report. Many additional recommendations were received from U.S. Nuclear Regulatory Commission (NRC) staff and external stakeholders, including the Office of Science and Technology Policy, Congress, international counterparts, other Federal and State agencies, nongovernmental organizations, the public, and the nuclear industry. These issues were raised in a variety of forums, including the staff's August 31, 2011, public meeting and a September 9, 2011, Commission meeting.

During its review of the NTTF recommendations, the Advisory Committee on Reactor Safeguards (ACRS) noted that Section 4.2 of the NTTF report discusses how the Fukushima operators faced significant challenges in understanding the condition of the reactors, containments, and spent fuel pools (SFPs) because the existing design-basis instrumentation was either lacking electrical power or was providing erroneous readings. As a result, an additional recommendation was developed to address the regulatory basis for requiring reactor and containment instrumentation to be enhanced to withstand beyond-design-basis accident conditions. This activity was prioritized as Tier 3 because it required further staff study and depended on the outcome of other lessons learned activities. The program plan for this recommendation was detailed in SECY-12-0095, "Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012 (ADAMS Accession No. ML12208A210).

Prior to the events at Fukushima Dai-ichi, the NRC had established requirements and guidance relative to assisting control room operators in preventing and mitigating the consequences of a reactor accident. The agency implemented and updated these requirements and guidance documents based on lessons learned from the 1979 accident at Three Mile Island Nuclear Station Unit 2 (TMI), severe accident policy decisions in the 1980s and 1990s, and enhancements made to nuclear power plants in response to the September 11, 2001, terrorist attacks.

As a result of the TMI accident, a set of generic safety issues was identified, including TMI Action Plan Item II.F.3, "Instruments for Monitoring Accident Conditions." The resolution of this item can be found in NUREG-0933, "Resolution of Generic Safety Issues," at http://nureg.nrc.gov/sr0933, which TMI Action Plan Item II.F.3 addressed several concerns regarding the availability and adequacy of instrumentation to monitor plant variables and systems during and following an accident. This item was resolved by establishing new requirements as described in a December 17, 1982, letter to all licensees of operating reactors, applicants for operating reactors, and holders of construction permits (ADAMS Accession No.

ML031080548). Regulatory Guide (RG) 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” Revision 2, included an expanded list of parameters and instrument ranges for licensees to consider to demonstrate that they met the underlying NRC requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria” (GDC), GDC 13, “Instrumentation and Control”; GDC 19, “Control Room”; and GDC 64, “Monitoring Radioactivity Releases.”

The industry developed severe accident management guidelines (SAMGs) in the 1990s to provide strategies and guidelines to mitigate the consequences of a severe accident. SECY-15-0065, “Proposed Rulemaking: Mitigation of Beyond-Design-Basis Events (RIN 3150-AJ49),” released May 15, 2015 (ADAMS Accession No. ML15049A201), discusses the history of the development of SAMGs. Enclosure 3 to SECY-15-0065, Section A.2, “Backfit Analysis of Rule Provisions that Constitute Backfits,” discusses how the Commission’s 1985 Severe Accident Policy Statement (50 Federal Register (FR) 32138) led to SAMGs being implemented at licensee facilities on a voluntary basis by the end of 1998. When it is determined that adequate core cooling is no longer assured, the licensee exits the plant emergency operating procedures (EOPs) or other governing processes and enters the SAMGs. The SAMGs are symptom-based, preplanned accident mitigation strategies that were developed using modern thermal-hydraulic and accident progression and consequence modeling. The SAMGs were developed for use in specific reactor designs and then adapted by individual licensees to reflect plant-specific design features and capabilities.

Following the events of September 11, 2001, the NRC issued orders that were eventually made generically-applicable via rulemaking, including 10 CFR Section 50.54(hh)(2), which requires licensees to develop and implement guidance and strategies 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 fires. These strategies can be found in extensive damage mitigation guidelines (EDMGs), which have been established at all U.S. operating nuclear power plants. The EDMGs are intended to be used when the normal command and control structure is disabled and the use of EOPs is not feasible. The development of EDMGs provides additional mitigation capabilities for beyond-design-basis accidents.

**Current Status**

The NRC staff has completed its assessment of this recommendation. As discussed below, the staff has determined that additional studies are unlikely to support potential regulatory requirements related to enhanced reactor and containment instrumentation for beyond-design-basis conditions that would be warranted when evaluated against 10 CFR Section 50.109, “Backfitting,” criteria for operating reactors or the issue finality provisions of 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” Although not needed to resolve the post-Fukushima Tier 3 item, the staff will continue participating in standard development organizations and updating regulatory guidance documents on the subject as part of its routine activities. For example, based on efforts by the Institute of Electrical and Electronics Engineers (IEEE) to provide guidance to address enhanced reactor and

containment instrumentation for beyond-design-basis conditions[3], the NRC staff plans to update and provide guidance in RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," for such instrumentation. If licensees of currently operating reactors so choose, they can use the revised guidance found in the future revision of RG 1.97 to enhance their reactor and containment instrumentation on a voluntary basis.

**Discussion**

The NRC staff's assessment of this recommendation considered recent Commission decisions that directly or indirectly affected the NRC staff's evaluation, including the work performed to support the Mitigation of Beyond-Design-Basis Events (MBDBE) rulemaking and the work associated with Fukushima-related orders. The assessment also considers Commission decisions associated with the review of new reactor severe accident instrumentation issues. The staff's assessment also includes an evaluation of the regulatory basis for enhanced capabilities for severe accident instrumentation, considering a review of instrumentation needs for implementing specific Commission-directed actions, previous and ongoing research efforts associated with severe accidents, and whether or not requirements to upgrade some instrumentation for operating reactors to withstand beyond-design-basis environments are needed.

*Commission Decisions Considered in the NRC Staff's Review of Enhanced Capabilities for Severe Accident Instrumentation*

This section of the staff's evaluation discusses recent Commission decisions for operating reactors that NRC staff considered during its review of this recommendation, along with past Commission decisions related to reviews of new reactors.

Operating Reactors

Mitigation of Beyond-Design-Basis Events Rulemaking

The Mitigation of Beyond-Design-Basis Events rulemaking makes generically applicable the requirements of Orders EA-12-049 and EA-12-051. These orders, as discussed later, include requirements associated with instrumentation.

As noted above, SECY-15-0065 provides a discussion regarding SAMGs and the staff's proposal that SAMGs be imposed as a regulatory requirement. In SECY-15-0065, the NRC staff did not propose additional requirements associated with instrumentation relied upon in SAMGs. SECY-15-0065, Enclosure 2, provides the following discussion regarding instrumentation used to support the SAMGs:

[3] IEEE is in the process of considering a revision to Standard 497, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." A draft version of IEEE Standard 497 includes a proposed revision that would broaden the scope of this standard to include severe accidents. This draft standard defines severe accidents as a subset of design extension conditions during which fuel damage has occurred.

> Specifically with regard to instrumentation relied upon in SAMGs, this rulemaking proposes no new permanent instruments beyond those required by Order EA-12-051 for spent fuel pools. The principles underlying this rulemaking recognize that it is not possible to design instrumentation that can directly measure plant parameters in all potential severe accident environments. As such, implementation of SAMG requirements in this framework would 1) provide for the use of alternate means for determining plant conditions when the primary means becomes unavailable or unreliable, 2) include courses of action to follow when the event degrades to the point where there is no reliable instrumentation available, 3) include consideration of potential uncertainties in instrumentation readings caused by anticipated severe accident environmental conditions, and 4) provide for the use of computational aids when direct diagnosis of key plant conditions cannot be determined safely from instrumentation. Finally, implementation of the proposed SAMG requirements would include the use of best estimate assumptions and calculations to determine operator actions as well as decision-making limits and action levels. Additionally, the Electric Power Research Institute (EPRI) has developed a technical basis document, the TBR [Technical Basis Report], for SAMGs which provides extensive technical basis information for this approach and information related to plant status assessment including conditions where some instrumentation may be unreliable or unavailable and provides alternatives for determining the strategy to use.

The Commission ultimately disapproved the imposition of SAMGs as a requirement in the SRM to SECY-15-0065, dated August 27, 2015 (ADAMS Accession No. ML15239A767) based on licensee commitments to implement and maintain SAMGs and their future inclusion in the Reactor Oversight Program.

Order EA-12-049 – Mitigating Strategies for Beyond-Design-Basis External Events

Order EA-12-049, “Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events,” dated March 12, 2012 (ADAMS Accession No. ML12054A735), contains requirements that have implications regarding additional capabilities to monitor accidents prior to the onset of core damage at nuclear power plants. The ACRS letter dated October 13, 2011 (ADAMS Accession No. ML11284A136), related to reactor and containment instrumentation enhancements, notes that immediately after the tsunami flooded the Fukushima Dai-ichi plant, key instrumentation for the reactor vessel, drywell, and wetwell were unavailable for Units 1 and 2 due to loss of alternating current (ac) and direct current power sources; the instruments at Unit 3 lost power nearly 30 hours later. When power was restored, reactor and containment conditions resulting from core damage had already deteriorated such that the validity of data from available sensors was questionable.

In response to Order EA-12-049, licensees are implementing requirements to ensure that instrumentation used to support the MBDBE strategies provide plant operators with information needed to implement core cooling and containment heat removal strategies prior to the onset of core damage and that such instrumentation remains powered during an extended loss of ac power (ELAP). The instrumentation is powered by safety-related batteries initially in the event of an ELAP, and by either onsite or offsite (i.e., FLEX equipment) power supplies to provide coping capabilities for an indefinite period of time.

The minimum set of parameters necessary to support the FLEX strategy is discussed in Section 3.2.1.10 of Nuclear Energy Institute (NEI) 12-06, Revision 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," issued May 2012 (ADAMS Accession No. ML12242A378). The NRC endorsed NEI 12-06, with clarification, in Japan Lessons-Learned Directorate (JLD) Interim Staff Guidance (ISG) JLD-ISG-2012-01, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated August 29, 2012 (ADAMS Accession No. ML12229A174). NEI 12-06, Section 3.2.1.10, states the following:

> In order to extend battery life, a minimum set of parameters necessary to support strategy implementation should be defined. The parameters selected must be able to demonstrate the success of the strategies at maintaining the key safety functions as well as indicate imminent or actual core damage to facilitate a decision to manage the response to the event within the Emergency Operating Procedures and FLEX Support Guidelines or within the SAMGs. Typically, these parameters would include the following:

| PWRs [Pressurized Water Reactors] | BWRs [Boiling Water Reactors] |
|---|---|
| • SG [Steam Generator] Level<br>• SG Pressure<br>• RCS [Reactor Coolant System] Pressure<br>• RCS Temperature<br>• Containment Pressure<br>• SFP Level | • RPV [Reactor Pressure Vessel] Level<br>• RPV Pressure<br>• Containment Pressure<br>• Suppression Pool Level<br>• Suppression Pool Temperature<br>• SFP Level |

> The plant-specific evaluation may identify additional parameters that are needed in order to support key actions identified in the plant procedures/guidance (e.g., isolation condenser (IC) level), or to indicate imminent or actual core damage.

In addition, the implementing guidance for Order EA-12-049 and the draft guidance for the proposed MBDBE rule address contingencies for the loss of all ac power. This includes taking local manual control of a (non-ac powered) pump, such as a turbine-driven auxiliary feedwater or reactor core isolation coolant pump and in support of this local manual action providing a mechanism for using a portable instrument capability (e.g., Fluke meter) that does not rely on the functioning of intervening electrical equipment.

Providing additional power sources or alternate means of monitoring this instrumentation throughout an accident's progression should aid licensee's understanding of the condition of the reactor vessel, containment, and spent fuel pools prior to instrumentation becoming unavailable or unreliable because of severe accident environmental conditions. This should therefore allow licensee's to more easily transition to the use of computational aids when direct diagnosis of key plant conditions cannot be determined safely from instrumentation.

Order EA-12-051 – Spent Fuel Pool Instrumentation

Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool

Instrumentation," dated March 12, 2012 (ADAMS Accession No. ML12056A044), requires nuclear power plants to install water level instrumentation in their spent fuel pools that must remotely report three distinct water levels: (1) normal level, (2) low level but still enough to shield workers above the pools from radiation, and (3) a level near the top of the spent fuel rods where more water should be added without delay. Order EA-12-051 contains requirements regarding the instrumentation's ability to provide reliable readings at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period of time. Section 3.4 of NEI 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051," issued August 2012 (ADAMS Accession No. ML12240A307), provides expectations for the qualification of the instrumentation. The NRC staff endorsed the guidance found in NEI 12-02, with exceptions and clarifications, in JLD-ISG-12-03, "Compliance with Order EA-12-051 Reliable Spent Fuel Pool Instrumentation," dated August 29, 2012 (ADAMS Accession No. ML12221A339).

Order EA-13-109 - Containment Vent Order

The NRC staff notes that compliance with Order EA-13-109, "Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," dated June 6, 2013 (ADAMS Accession No. ML13143A321), involves severe accident capable containment instrumentation requirements for Mark I and II containments. Order EA-13-109 instrumentation requirements are discussed below to provide additional information on instrumentation requirements that were identified as a result of Fukushima lessons learned.

The NRC staff guidance regarding severe accident capable instrumentation associated with the this order can be found in JLD-ISG-2015-01, "Compliance with Phase 2 of Order EA-13-109, Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," issued April 2015 (ADAMS Accession No. ML15104A118). This ISG endorses, with exceptions and clarifications, the methods described in the industry guidance document NEI 13-02, "Industry Guidance for Compliance with Order EA-13-109," Revision 1, dated April 23, 2015 (ADAMS Accession No. ML15113B318). NEI 13-02 notes that instrumentation needed to support severe accident water addition (SAWA) or severe accident water management (SAWM) is normally powered by safety-related power sources that are expected to be repowered by FLEX portable equipment and procedures, such that functionality is continuously maintained. The difference between FLEX and SAWA/SAWM is that the capability must be demonstrated to power the instruments under severe accident conditions. Additional details concerning SAWA and SAWM instrumentation are contained in NEI 13-02, Sections 4 and 5 and Appendices C and I.

NEI 13-02, Section 4.2.4.2, states the following:

> The means to monitor system status should support Sustained Operations during an ELAP, and be designed to operate under environmental conditions that would be expected following a loss of containment heat removal capability and an ELAP.

<u>New Reactors</u>

The Commission's Severe Accident Policy Statement, issued in 1985 (50 FR 32138),

documents the Commission's determination that for existing reactors, severe accidents must pose no undue risk to public health and safety. However, the Commission noted that this determination for existing reactors should not be viewed as implying that safety improvements in new plant designs should not be actively sought. The Commission further stated that it fully expects that vendors engaged in designing new plants will achieve a higher standard of severe accident safety performance than prior designs. This policy led to the development of criteria for instrumentation enhancements for new reactors.

For new reactors the applicable criteria for equipment, both electrical and mechanical, required to mitigate the consequences of ex-vessel severe accidents is discussed in Section III.F, "Equipment Survivability," of SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements," dated January 12, 1990 (ADAMS Accession No. ML003707849). The NRC staff provided further guidance for new reactor severe accident instrumentation in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993 (ADAMS Accession No. ML003708021). The Commission approved the positions regarding equipment survivability for new reactors in SRMs dated June 26, 1990, and July 21, 1993, for SECY-90-016 and SECY-93-087, respectively. SECY-93-087 states that equipment provided only for severe accident protection need not be subject to the equipment qualification requirements in 10 CFR 50.49, "Equipment Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"; the quality assurance requirements in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; or the redundancy and diversity requirements in 10 CFR Part 50, Appendix A. However, mitigation features must be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed.

The expectation that new reactors will address equipment survivability can be found in the following documents:

- Regulatory Position C.I.19.8, "Severe Accidents," of RG 1.206, "Combined License Applications for Nuclear Power Plants," (ADAMS Accession No. ML070630023)

- Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," Revision 2 of NUREG-0800, "Standard Review Plan" (ADAMS Accession No. ML071700652)

For example, the AP1000 equipment survivability assessment includes the following methodology to demonstrate equipment survivability:

- Identify the high-level actions used to achieve a controlled, stable state.

- Define the accident time frames for each high level action.

- Determine the equipment and instruments used to diagnose, perform, and verify high level actions in each time frame.

- Determine the bounding environment within each time frame.

- Demonstrate reasonable assurance that the equipment will survive to perform its function within the severe environment.

*NRC Process Used to Evaluate the Regulatory Basis for Enhanced Capabilities for Severe Accident Instrumentation*

The NRC staff's process for evaluating the regulatory basis for enhanced capabilities for severe accident instrumentation divided the activities for this issue into the following three tasks:

1. Ensure that licensees and NRC staff are appropriately considering instrumentation needs when implementing site-specific actions (e.g., related to post-Fukushima regulatory actions).

2. Obtain and review information from previous and ongoing research efforts for severe accident management analysis. This task also involved coordination with international and domestic entities.

3. Evaluate the results of Tier 1 activities in coordination with the information obtained from applicable research efforts (international and domestic) to determine if possible requirements for enhanced instrumentation are needed.

<u>Task 1</u>

The MBDBE rulemaking activities capture issues associated with NTTF Recommendations 4.1 and 8, and make generically applicable requirements associated with Order EA-12-049 and EA-12-051. The NRC staff associated with the review of this issue were also involved in the development of the following guidance documents associated with the MBDBE rulemaking: DG-1301, "Flexible Mitigation Strategies for Beyond-Design-Basis Events;" ," issued April 2015, and DG-1317, "Wide-Range Spent Fuel Pool Level Instrumentation;" and DG-1319, "Integrated Response Capabilities for Beyond-Design-Basis Events," both issued March 2015. In addition, the NRC staff associated with the review of the issue provided comments on SAMG instrumentation found in the MBDBE proposed rule that was provided to the Commission on April 30, 2015. As discussed in the regulatory analysis that accompanies the proposed rule, the NRC staff proposed a requirement associated with SAMGs. SECY-15-0065, states the following regarding SAMGs:

> Because the available quantitative risk information is not a complete measure of the SAMG safety benefits, the staff relied on quantitative and qualitative reasons to conclude that SAMG requirements would result in substantial additional protection for public health and safety, as stated in Appendix A to the supporting draft regulatory analysis. Specifically, quantitative risk information indicates that SAMGs have a small safety benefit. In addition, SAMGs directly support maintenance of containment integrity

> following severe accidents and indirectly support the protective action recommendations made by the emergency response organization in such circumstances and as such, the SAMGs have a very important link to two foundational parts of the NRC's defense-in-depth framework: containment and emergency preparedness.

The Commission disapproved the staff's proposal in its SRM dated August 27, 2015, and concluded that imposing SAMGs requirements (as opposed to continuing of the industry voluntary initiative) does not meet the backfitting requirements of 10 CFR 50.109. As part of the staff's proposal to include requirements for SAMGs, the staff concluded that SAMG instrumentation should not be part of the proposed SAMG requirement, in view of the quantified risk insights. The NRC staff's assessment concludes that in light of these risk insights and the Commission's decision in its SRM, SAMG instrumentation requirements would also not meet the backfitting requirements of 10 CFR 50.109.

As discussed above, the NRC staff also considered Tier 1 activities related to instrumentation requirements associated with Order EA-13-109. Guidance documents associated with the vent order include expectations related to power availability and the environmental conditions expected with a loss of containment heat removal and an ELAP.

Because the NRC team that was assigned to evaluate the regulatory basis for enhancements to severe accident instrumentation has been, and continues to be, involved with the Tier 1 issues described above, the NRC staff has confidence that it and licensees are appropriately considering instrumentation needs when implementing site-specific actions for Tier 1 activities associated with the MBDBE rulemaking and Order EA-13-109.

Task 2

In accordance with Task 2, the NRC staff has been actively engaged with a number of domestic and international organizations.

It should be noted that the NRC performs severe accident research in partnership with nuclear safety agencies and institutes in more than 20 countries. In addition, the NRC staff continues to engage the U.S. Department of Energy (DOE) and various trade organizations to ensure that Fukushima lessons-learned related to instrumentation capabilities during a severe accident are appropriately considered. Although the staff's assessment concludes that new regulatory requirements to enhance capabilities for severe accident instrumentation do not pass the backfitting analysis criteria for operating reactors, the NRC staff plans to continue to engage in severe accident research activities with international and national organizations associated with the capabilities of instruments to withstand severe accident environments. The outcome of such research will, in part, support ongoing efforts to upgrade SAMGs so that nuclear power plant operators understand the instrumentation limitations associated with SAMGs. A better understanding of instrumentation limitations in a severe accident environment has the potential for enhancing SAMGs, and thus an operator's ability to prevent or mitigate severe accidents.

A list of significant activities under Task 2 is summarized below:

1. International Atomic Energy Agency (IAEA)

    - IAEA Nuclear Energy Series No. NP-T-3.16, “Accident Monitoring Systems for Nuclear Power Plants,” February 2015

    - New Working Group – Instrumentation and Control Equipment Qualification Best Practices

    - IAEA Safety Standards Series No. NS-G-2.15, “Severe Accident Management Programmes for Nuclear Power Plants,” Vienna, 2009

2. Organization for Economic Cooperation and Development/Nuclear Energy Agency (NEA)

    - Report of the Committee on Nuclear Regulatory Activities (CNRA) Task Group on Accident Management, NEA/CNRA/R(2014)2, “Accident Management Insights after the Fukushima Daiichi NPP Accident”

3. Multinational Design Evaluation Program

    - Evolutionary Pressurized Water Reactor Technical Experts Subgroup for Severe Accidents

4. EPRI

    - EPRI Technical Report TR-1025295, “Severe Accident Management Guidance Technical Basis Report,” 2012

    - EPRI Technical Report TR-1026539, “Investigation of Strategies for Mitigating Radiological Releases in Severe Accidents; BWR Mark I and Mark II Studies,” September 2012

    - New EPRI Project, “Instrumentation & Control for Beyond-Design-Basis Events and Severe Accidents”

5. DOE

    - Sandia National Laboratories, Sandia Report, SAND2012-6173, “Fukushima Daiichi Accident Study” (status as of April 2012), August 2012

    - Idaho National Laboratory report INL/EXT-13-28043, “TMI-2 - A Case Study for PWR Instrumentation Performance during a Severe Accident,” March 2013

    - Oak Ridge National Laboratory report ORNL/TM-2013/154, “Fukushima Daiichi – A Case Study for BWR Instrumentation and Control Systems Performance during

a Severe Accident," April 2013

- Collaboration in a Japanese study on instrumentation performance at Fukushima
- Plant specific studies on severe accident instrumentation needs and performance
    - Draft report ORNL/TM-2015/278, "Post-Severe Accident Environmental Conditions for Essential Instrumentation for Boiling Water Reactors"
    - Draft report INL/EXT-15-35940. "Scoping Study Investigating PWR Instrumentation during a Severe Accident Scenario"

6. National Academy of Sciences (NAS) report
    - "Lessons Learned from the Fukushima Nuclear Accident for Improving Safety of U.S. Nuclear Plants," 2014
7. Interface with Standards Development Organization
    - The NRC staff plans to Update RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," based on the planned new revision of IEEE Standard 497, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations"

<u>Task 3</u>

As part of Task 3, the NRC staff used the factors discussed above to determine if requirements for enhanced severe accident instrumentation could be justified under the NRC's regulatory framework. The NRC staff review was performed considering the Commission's August 27, 2015, SRM that disapproved the imposition of SAMGs as a requirement. Although SAMGs are not a regulatory requirement, they are being voluntarily upgraded in response to Fukushima lessons learned, and the NRC staff is assessing these upgrades. The NRC staff's evaluation also included a review of instrumentation relied upon for the MBDBE rulemaking and the instrumentation relied upon for SAMGs. Further, the NRC staff considered ongoing work on IEEE Standard 497. The discussion that follows provides the results of the staff's evaluation.

Operating Reactors

Using the insights described above, the staff has determined that there is little likelihood that further study or research would enable the NRC to recommend additional requirements for licensees to enhance reactor and containment instrumentation to support monitoring capability during severe accidents. Based largely on the analyses completed for the MBDBE rulemaking, the staff has concluded that the imposition of such a regulatory requirement would not represent a substantial safety benefit to public health and safety. As a result, enhanced reactor and containment instrumentation requirements are unlikely to satisfy the criteria in 10 CFR 50.109(a)(3) for backfitting an operating reactor. The NRC staff's determination is also

based on consideration of the substantial safety improvements already being implemented as part of NRC's post-Fukushima regulatory actions, such as Order EA-12-049, Order EA-13-109, the MBDBE rulemaking, and voluntary industry initiatives.

Quantified Risk Information

Enclosure 3 of SECY-15-0065 (ADAMS Accession No. ML15049A212), Section A.2, "Backfit Analysis of Rule Provisions that Constitute Backfits," discusses of quantified risk information as it relates to imposing SAMGs as a requirement in the MBDBE rulemaking. The document provides estimates of the risk of latent cancer fatality compared against the Commission's Safety Goal Policy quantitative health objective (QHO), which is a measure that equates to 1/10 of 1 percent of the individual latent cancer fatality risk. The quantitative metric for the individual latent cancer facility risk is approximately $2x10^{-6}$ per reactor year. The analysis concludes that SAMGs would have a small safety benefit and would not significantly help plants maintain margin to the QHO. The conclusion is based on the risk of a severe accident being low and that existing emergency preparedness requirements ensure that the surrounding population is adequately protected in the unlikely event a severe accident occurs. The staff notes that, given that SAMGs could not be justified based on quantified risk information alone, the imposition of enhanced reactor and containment instrumentation requirements to further improve SAMGs are similarly not justified based on risk.

Additional Considerations

The SAMG backfit analysis supporting SECY-15-0065 provided a qualitative analysis proposing the imposition of SAMGs as a requirement because it would reflect a defense-in-depth philosophy and would support actions and decisions to:

1. Halt the progression of the accident (if possible).

2. Minimize or delay the release of fission products (including making best use of the containment.)

3. Cope with the radiological conditions, make decisions regarding onsite mitigation, notify offsite organizations, and make recommendations regarding offsite protective actions.

As discussed above, the Commission, in its SRM dated August 27, 2015 (ADAMS Accession No. ML15239A767), disapproved the staff's recommendation that SAMGs be imposed as a requirement. Regardless, even if the Commission had approved the staff's proposal, SECY-15-0065 states that new instrumentation requirements would not be imposed as part of the rulemaking and would need to be justified on their own merits. The SAMGs were developed and implemented based on a philosophy that makes use of available instrumentation, includes backup or alternative means for determining plant conditions when the primary means become unavailable or unreliable, and includes a course of action to follow when the event degrades to the point where there is no reliable instrumentation available.

The NRC staff compared the approach described in IAEA Report NP-T-3.16 to U.S. approaches used as part of compliance with Order EA-12-049 and in the SAMGs. Specifically, annex I to IAEA Report NP-T-3.16 includes a detailed discussion of the SA-Keisou approach being used in

Japan (in English SA-Keisou means severe accident instrumentation and monitoring systems). The NRC staff reviewed several licensees' mitigating strategies developed under Order EA-12-049 to identify the list of required instrumentation and compared those instruments with the SA-Keisou list. The staff found various differences between these lists, depending on the specific reactor type and plant. On the other hand, Boiling Water Reactor Owners Group (BWROG) and Westinghouse Owners Group (WOG) SAMG instrumentation is similar to the Japanese SA-Keisou instrumentation (with some exceptions). The exceptions are limited in nature (e.g., the SA-Keisou instrumentation includes BWR drywell water level and PWR reactor cavity level instrumentation that are not found in the SAMGs).

The NRC staff closely reviewed the purpose, assumptions, approaches, and considerations for the SA-Keisou approach versus the approach used for Order EA-12-049 compliance. The SA-Keisou approach assumes worst-case, severe accident conditions, while the mitigating strategies approach assumes an ELAP and loss of normal access to the ultimate heat sink, but not core damage. The mitigating strategies parameters were selected to demonstrate the success of the strategies at maintaining key safety functions, and to indicate imminent or actual core damage to facilitate decision making and event management. The mitigating strategies parameters also assume that station battery life is extended by providing power to only the minimum set of parameters necessary to support strategy implementation.

Because of the different assumptions and objectives of Order EA-12-049 versus the SA-Keisou approach, it is not unexpected that the list of instrumentation relied upon would be different. However, the NRC staff notes that the BWROG and WOG list of SAMG instrumentation generally aligns with the SA-Keisou approach, although as discussed above, some exceptions were identified. Regarding these exceptions, the NRC staff notes that as discussed in SECY-15-0065, SAMGs are expected to provide for the use of computational aides when direct diagnosis of key plant conditions cannot be determined reliably from installed instrumentation.

The SA-Keisou methodology also assumes that instrumentation and monitoring systems are to be designed to have environmental resistance, including resistance to temperature, pressure, humidity, and radiation conditions associated with a severe accident. In SECY-15-0065, the staff noted that an updated SAMG framework would include consideration of potential uncertainties in instrumentation readings caused by anticipated severe accident environmental conditions. To this end, the NRC staff is aware that some licensees are considering the use of simulators that include the ability to model severe accident conditions using software modules based on the MELCOR code. By modeling severe accident conditions using MELCOR, the simulators can model certain accident progression scenarios through the design-basis environment in which instrumentation is expected to be reliable. While some licensees appear to be developing these severe accident simulators separate from the control room simulators, the NRC is aware that one licensee is updating its control room simulator with a detailed flooding model for both internal and external flooding that dynamically simulates what equipment and access will be lost as the water level rises. However, it is not clear whether the instrumentation system performance under the environmental conditions resulting from the MELCOR-developed scenarios has been modeled with sufficient fidelity to represent expected performance under such environmental conditions.

The NRC staff notes that part of the ACRS's concern, which led to the recommendation regarding enhanced instrumentation for severe accidents, was that Fukushima Dai-ichi

operators faced challenges in understanding the condition of the reactors, containments, and SFPs because the existing design-basis instrumentation was either lacking electrical power or providing erroneous readings. Regarding electrical power for instrumentation, as discussed above, in response to Order EA-12-049 and the proposed MBDBE rulemaking, licensees are implementing strategies to ensure that instrumentation needed to comply with these requirements remains powered during an ELAP. These actions will ensure that the minimum set of instrumentation necessary to implement the mitigating strategies should remain powered throughout the event, providing the parameters necessary to demonstrate maintenance of key safety functions, as well as indicate imminent or actual core damage. This should also aid a licensee's understanding of the condition of the reactor vessel, containment, and SFPs prior to instrumentation becoming unavailable or unreliable because of severe accident environmental conditions, which would allow licensees to more easily transition to the use of computational aids when direct diagnosis of key plant conditions cannot be determined safely from instrumentation.

As described above, the NRC staff believes that it is necessary for the NRC to continue to engage in severe accident research activities associated with the capabilities of instruments to withstand severe accident environments. The outcome of such research will aid industry efforts to upgrade SAMGs so that nuclear power plant operators understand the limitations associated with the instrumentation relied on to implement SAMGs. A better understanding of instrumentation limitations in a severe accident environment has the potential for enhancing SAMGs and thus an operator's ability to prevent, or mitigate severe accidents.

Finally, the NRC staff also notes that a revision to IEEE Standard 497 is planned. Members of the NRC staff have participated in the development of the next revision to IEEE Standard 497, which was influenced by the work found in IAEA Report NP-T-3.16. The NRC staff is planning to update RG 1.97 to address the revision to IEEE Standard 497, so that licensees of currently operating reactors may voluntarily choose to use the revised guidance found in the future revision of RG 1.97 to enhance their reactor and containment instrumentation.

New Reactors

In accordance with Commission policy established in the 1990s, designs that have been certified in accordance with 10 CFR Part 52, Subpart B, "Standard Design Certifications," have been analyzed for equipment survivability. The equipment survivability analysis provides reasonable assurance that equipment for severe accident protection will operate in the severe accident environment for which it is intended and over the time span for which it is needed. Since this policy was in place prior to any design being certified, it does not constitute a backfit and is consistent with the finality provisions found in 10 CFR Part 52.

Regarding imposition of provisions in Order EA-12-049 related to providing power to instrumentation needed to implement MBDBE strategies should an ELAP occur, any backfitting and finality issues were addressed as part of the issuance of the orders. Therefore, for new reactors, in addition to the equipment survivability analysis discussed above, the staff notes that as a result of Order EA-12-049, strategies will be implemented to ensure mitigating strategies instrumentation will remain powered during an ELAP.

Summary of Staff's Assessment

For operating reactors, recent studies on the expected frequency of severe accidents and the ability to take protective actions (e.g., evacuations) have determined that while enhancements to instrumentation or other activities related to severe accident management might provide marginal safety improvements, they are not needed for operating plants to meet the QHOs and they do not represent a substantial safety improvement as would be required to impose additional regulatory requirements.

For new reactors, the Commission policy decisions in the 1990s resulted in equipment survivability evaluations that have been, and will continue, to be performed to provide reasonable assurance that the equipment provided for severe accident protection will operate in the severe accident environment for which it is intended and over the time span for which it will be needed.

For both operating and new reactors, enhancements to the power supplies for mitigating strategies instrumentation have been, or will be, implemented in response to Order EA-12-049 and its associated rulemaking. This limited set of instrumentation provides the parameters necessary to demonstrate the success of the strategies at maintaining key safety functions, as well as indicating imminent or actual core damage to facilitate a decision to manage the response to the event within the emergency operating procedures and FLEX support guidelines or within the SAMGs. Providing additional power sources to this instrumentation throughout an accident's progression should aid licensee's understanding of the condition of the reactor vessel, containment, and SFPs prior to instrumentation becoming unavailable or unreliable under severe accident environmental conditions.

**Stakeholder Interactions**

The NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015.

The NRC staff intends to discuss the recommendation to update RG 1.97 to provide guidance for enhanced reactor and containment instrumentation for beyond-design-basis events in a Category 2 public meeting. The NRC staff also intends to brief the ACRS Fukushima subcommittee and, if necessary, the ACRS full committee.

**Conclusion**

Based on the evaluation described above, the staff does not expect that further regulatory action is needed to close this recommendation. However, the staff plans to interact with the ACRS and external stakeholders and provide more detailed documentation, incorporating insights from these interactions, to the Commission by March 2016.

Regarding the initiative to update RG 1.97, the update will include the revision to IEEE Standard 497 and will take approximately 1 year after the revision to IEEE Standard 497 is

issued. As discussed above the revision to IEEE Standard 497 is scheduled for completion in early Calendar Year 2016. If licensees of currently operating reactors so choose, they can use the guidance found in the revision of RG 1.97 to enhance their reactor and containment instrumentation on a voluntary basis. New reactors will continue to assess equipment survivability for reactor and containment instrumentation for beyond-design-basis events, in accordance with Commission policy.

**Enclosure 6: Proposed Resolution Plan for Basis of Emergency Planning Zone Size and Pre-Staging of Potassium Iodine Beyond 10 Miles**

**Background**

As directed by staff requirements memorandum to SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated August 19, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112310021), the staff sought to identify additional recommendations related to lessons learned from the Fukushima Dai-ichi event, beyond those identified in the Near-Term Task Force (NTTF) report. Many additional recommendations were received from U.S. Nuclear Regulatory Commission (NRC) staff and external stakeholders, including the Office of Science and Technology Policy, Congress, international counterparts, other Federal and State agencies, nongovernmental organizations, the public, and the nuclear industry. These issues were raised in a variety of forums, including the staff's August 31, 2011, public meeting and a September 9, 2011,, Commission meeting.

As part of its evaluation of additional recommendations, the staff identified a Tier 3 recommendation to reconsider the basis of the emergency planning zone (EPZ) size and practices associated with the prestaging of potassium iodide (KI) beyond 10 miles (19.0394 km). This was determined to be a Tier 3 issue because further assessment and information from the Fukushima accident were needed before the evaluation could be completed.

Related to this issue, a petition for rulemaking (PRM) (PRM-50-104) was filed on February 15, 2012 (ADAMS Accession No. ML120488004). The petitioner requested that the Commission amend its regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," to expand existing EPZs around nuclear power plants, create a new EPZ, and require the incorporation of concurrent natural disasters in the required periodic emergency plan drills. The NRC published a notice of acceptance, docketing, and request for public comments in the *Federal Register* (FR) on April 30, 2012 (77 FR 25375). The comment period for this PRM closed on July 16, 2012. The NRC received 5,993 comment submissions, 5,953 of which supported the petition and 40 of which opposed the petition. Of the 5,993 comment submissions, 5,702 were form letters and 291 were unique submissions. The NRC prepared a comment response document to demonstrate how all comments were considered and to respond to the issues identified in the comments. The comment response document is available in ADAMS under Accession No. ML14042A227.

The NRC staff provided its recommendation on this PRM to the Commission in SECY-13-0135, "Denial of Petition for Rulemaking Requesting Amendments Regarding Emergency Planning Zone Size (PRM-50-104)," dated February 27, 2014 (ADAMS Accession No. ML13109A503). The NRC staff concluded that the current size of EPZs is appropriate for existing reactors and that emergency plans provide an adequate level of protection for public health and safety in the event of an accident at a nuclear power plant. The current EPZs provide for a comprehensive emergency planning framework that would allow expansion of the response efforts beyond the

designated distances should events warrant such an expansion. On February 27, 2014, the Commission approved the staff's recommendation to deny this PRM.

**Current Status**

The staff continues to conclude, after extensive reviews, that the current size of EPZs is appropriate for existing reactors and that emergency plans provide an adequate level of protection of the public health and safety in the event of an accident at a nuclear power plant. In addition, the staff concludes that the current distribution program for KI tablets within the 10 mile EPZ provides an adequate level of protection of public health and safety.

The staff maintains awareness of international scientific organizations (i.e., International Atomic Energy Agency (IAEA), the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), and the World Health Organization (WHO)), which continue to monitor the health and environmental impacts of the radioactive releases from the Fukushima Dai-ichi reactors.

**Discussion**

The NRC's regulations in 10 CFR Part 50 require two EPZs around each nuclear power plant. The 10-mile zone establishes the area in which exposure from a radiological release would likely occur and protective actions, such as sheltering in place or evacuation, would be appropriate. The 50-mile zone is the ingestion exposure pathway EPZ, where human exposure to radionuclides would likely result from ingestion of contaminated food, milk, or surface water. Nuclear power plant licensees, Federal, State, and local governments, and offsite response organizations perform comprehensive planning for these zones and routinely test and evaluate these plans through full participation exercises. The licensee develops the onsite emergency plan for NRC review. The State and local governments develop and maintain the offsite emergency plans, which are evaluated by the Federal Emergency Management Agency (FEMA). Through coordination of their emergency plans, the licensee and State and local governments establish the EPZ for the respective site.

Following the event at Fukushima Dai-ichi, the NRC, in conjunction with other U.S. Government entities, issued a conservative travel advisory for American citizens within a 50-mile range of the Fukushima site. The 50-mile travel advisory was made in the interest of protecting the health and safety of U.S. citizens in Japan based on the information available at that time and the rapidly evolving situation. Because of this action, the staff determined that it was appropriate to consider whether the basis of current EPZ requirements for U.S. nuclear power plants provides reasonable assurance of adequate protection of public health and safety. NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," issued November 1978, provides the technical basis for the plume exposure pathway EPZ and an ingestion exposure pathway EPZ. NUREG-0396 analyzes a spectrum of potential nuclear plant accidents and determines the size of EPZs, in which detailed planning would be appropriate for the protection of public health and safety.

The task force that developed NUREG-0396 considered several possible rationales for establishing the size of the EPZs, including risk, cost effectiveness, and the accident consequence spectrum. After reviewing these alternatives, the task force concluded that the objective of emergency response plans should be to provide dose savings for a spectrum of accidents that could produce offsite doses in excess of the U.S. Environmental Protection Agency's (EPA's) protective action guides (PAGs) (EPA-400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," issued May 1992). This rationale established bounds for the area in which detailed planning would be required as a defense-in-depth measure.

The EPZ requirements also provide consistency in nuclear plant preparedness across the nuclear fleet and the supporting State and local governments. All U.S. nuclear power plants currently have approved emergency plans that include EPZs in compliance with the regulations. FEMA provides oversight of offsite response plans that support nuclear power plants. Any changes to EPZs will be reviewed in coordination with FEMA.

Since the issuance of NUREG-0396, the staff has conducted several studies useful in evaluating the adequacy of the plume exposure pathway EPZ. NUREG/CR-6953, "Review of NUREG-0654, Supplement 3, 'Criteria for Protective Action Recommendations for Severe Accidents'" (ADAMS Accession Nos. ML080360602 (Vol. 1), ML083110406 (Vol. 2), and ML102380087 (Vol. 3)), evaluates the efficacy of various protective action strategies within the EPZ. NUREG/CR-6864, "Identification and Analysis of Factors Affecting Emergency Evacuations" (ADAMS Accession Nos. ML050250245 (Vol. 1) and ML050250219 (Vol. 2)), examines large evacuations in the U.S. between 1990 and 2003 to gain a fuller understanding of the dynamics involved. NUREG/CR-6981, "Assessment of Emergency Response Planning and Implementation for Large Scale Evacuations," issued March 2008 (ADAMS Accession No. ML082960499), assessed Hurricanes Katrina, Rita, and Wilma, as well as other large scale evacuations, for lessons learned to further enhance the emergency preparedness program for radiological emergencies at nuclear power plants. Evacuations related to these incidents have revealed issues that have not been previously encountered during large scale evacuations. The knowledge gained from studying 11 large scale evacuations (in excess of 10 million people) was used to determine if the emergency planning activities were effective in managing the response effort. The 11 incidents covered wide geographical areas and affected 14 nuclear power plant EPZs, although none of the evacuations were related to issues associated with those nuclear power plants. Research of evacuations such as these provided an opportunity to understand contributing factors that support the effectiveness of emergency response activities. A key finding of NUREG/CR-6981 is that emergency planning for nuclear power plants has substantially anticipated and addressed the issues identified in the large scale evacuations.

Draft NUREG-1935, "State-of-the-Art Reactor Consequence Analysis (SOARCA) Report" (ADAMS Accession No. ML120250406), evaluates hypothetical evacuations within EPZs and beyond, in response to a series of accident scenarios. These analyses informed the staff's conclusion that the current requirements for EPZs remain protective of public health and safety. In response to a frequently asked question regarding protective action recommendations, the staff informed all licensees that it is a regulatory requirement for a licensee to develop and communicate a protective action recommendation when EPA PAG doses may be exceeded beyond the 10-mile plume exposure pathway EPZ.

In addition, the staff has reviewed information obtained from health studies on the affected populations. These studies have been undertaken by the Japanese Government (Fukushima Health Management Survey), as well as the WHO and the UNSCEAR. The Fukushima Health Survey is a long-term study that will follow affected populations for years to assess radiological health impacts from low-dose radiation. These health studies based their conclusions, to date, on modeling, dose reconstruction, and real world information, such as field data and thyroid bioassays.

The Japanese government had arrangements for protective measures in place prior to the accident. These arrangements included criteria for sheltering, evacuation, and the use of potassium iodide for thyroid blocking. During the event the government evacuated and sheltered populations in the areas around the plant site. The use of potassium iodide was not implemented uniformly, primarily due to the lack of detailed arrangements. In addition, in some areas, the population was already evacuated when decisions regarding the use of potassium iodide were made.

The IAEA 2015 technical report on the accident at Fukushima attributes the low thyroid doses of children partly to restrictions placed on drinking water and food, including leafy vegetables and fresh milk. While dairy products were not the main pathway for the ingestion of radioiodine in Japan, it is clear that the most important method of limiting thyroid doses, especially to children, is to restrict the consumption of fresh milk from grazing cows.

UNSCEAR issued its scientific report in April 2014, which assessed radiation doses and associated effects on health and the environment from the Fukushima Dai-ichi accident. The UNSCEAR found that the radiation exposure of the Japanese population was low, leading to correspondingly low risks of health effects due to radiation later in life. The average effective doses for adults in evacuated and nonevacuated areas of the Fukushima Prefecture caused by the releases from the Fukushima reactors ranged from a few up to about 10 millisieverts (mSv). The effective doses for 10-year-old children and 1-year-old infants were estimated to be about twice as high. For neighboring prefectures and for the rest of Japan, doses were lower. To provide context, the average effective dose received annually in Japan from natural background radiation is about 2.1 mSv.

Average absorbed doses to the thyroid among those most exposed individuals ranged from up to about 35 milligray (mGy) for adults and up to about 80 mGy for a 1-year old. This is significantly higher than absorbed doses to the thyroid from natural background radiation; the average annual absorbed dose to the thyroid from naturally occurring sources of radiation is typically of the order of 1 mGy. The WHO, in its 1999 publication, “Guidelines for Iodine Prophylaxis Following Nuclear Accidents, Update 1999,” recommends a generic intervention level of 100 mGy avertable dose for all age groups, although it recommends that a lower intervention level be considered for small children. This recommendation for a much lower intervention level was based largely on the increased number of thyroid cancers seen in young children as a result of the Chernobyl accident. The U.S. Food and Drug Administration of the U.S. Department of Health and Human Services, issued its guidance on KI use in 2001, “Guidance Potassium Iodide as a Thyroid Blocking Agent in Radiation Emergencies.” The

recommended intervention level for children was 5 rem child thyroid dose, also based on those early reports of increased pediatric thyroid cancer.

Ongoing followup health studies undertaken by WHO and UNSCEAR have largely attributed the increased rates of childhood thyroid cancers from the Chernobyl accident to the high levels of radioactive iodine released during the accident and to the failure to interdict the milk supply, as well as an iodine insufficiency in the local diet. In its 2006 publication of health effects of the Chernobyl accident[4], WHO concluded that "since radioactive iodine is short lived, if people had stopped giving locally supplied contaminated milk to children for a few months following the accident, it is likely that most of the increase in radiation-induced thyroid cancer would not have resulted."

As a result of the Fukushima nuclear power plant accident, the Fukushima Prefecture launched the Fukushima Health Management Survey to investigate long-term, low-dose radiation exposure caused by the accident. The survey estimated radiation exposure from the accident and more detailed dose assessments by recreating the whereabouts of every Fukushima Prefecture resident for the 4 month period beginning on March 11, 2011, the date of the nuclear accident. The primary purposes of this survey are to monitor the long-term health of residents, promote their future well-being, and confirm whether long-term, low-dose radiation exposure has health effects. The June 2015 report of the Fukushima Health Management Survey[5] stated that doses have been estimated for approximately 449,000 survey respondents. Over 99.8 percent of the respondents received doses less than 5 mSv, while the highest dose was estimated to be 25 mSv.

To date, the results of these ongoing studies do not challenge the EPZ planning basis nor the KI distribution program. As such, the staff's assessment is that no further regulatory action is necessary in response to this recommendation and that it can be closed. The staff plans to continue to monitor the studies being conducted by WHO, UNSCEAR, and the Fukushima Health Management Survey, and engage stakeholders through the appropriate forums. Such forums include: the annual Regional State Liaison Officers meeting; the quarterly meeting of the Federal Radiological Preparedness Coordinating Committee, and the annual meeting of the Conference of Radiation Control Program Directors.

**Stakeholder Interactions**

The staff has had extensive stakeholder interactions on these recommendations as part of routine activities, post-Fukushima correspondence, and the resolution of PRM-50-104.

In addition, the NRC staff provided the Fukushima subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015.

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[4] An overview can be found at http://www.who.int/ionizing_radiation/chernobyl/backgrounder/en/ (accessed September 4, 2015)

[5] Available at http://fmu-global.jp/fukushima-health-management-survey/#report (accessed September 4, 2015)

**Conclusion**

Subject to Commission approval and based on the staff's assessment provided above, the staff is closing this recommendation.

**Enclosure 7: Proposed Resolution Plan for Tier 3 Recommendations 9, 10, and 11 - Emergency Preparedness Activities Not Addressed Elsewhere**

## Introduction

On July 13, 2012, the U.S. Nuclear Regulatory Commission (NRC) issued SECY-12-0095, "Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12165A092), in which the following Tier 3 items were included within one program plan:

- Emergency preparedness (EP) enhancements for prolonged station blackout (SBO) and multi-unit events;
- Emergency response data system (ERDS) capability;
- Additional EP topics for prolonged SBO and multi-unit events; and
- EP topics for decision making, radiation monitoring, and public education.

These items collectively originated from Near-Term Task Force (NTTF) Recommendations 9.1, 9.2, 9.3, 10.1, 10.2, 10.3, 11.1, 11.2, 11.3, and 11.4. The program plan outlined in SECY-12-0095 described an approach to address these items collectively using an advance notice of proposed rulemaking (ANPR). The NRC staff has since determined that it would be more appropriate to address the recommendations in smaller groups, with more tailored paths to resolution. Those proposed resolution plans are discussed in more detail in the following sections:

- Section I. Recommendations Subsumed into the Mitigation of Beyond-Design-Basis Events Rulemaking
- Section II. Resolution Plan for Recommendation 9.3 (Partial)
- Section III. Resolution Plan for Recommendation 10.3a, Alternative Method for Transmitting ERDS
- Section IV. Resolution Plan for Recommendation 10.3b, ERDS Data Set
- Section V. Resolution Plan for Recommendation 10.3c, ERDS Continuous Transmission
- Section VI. Resolution Plan for Recommendation 11.2, Evaluate Recovery and Reentry Insights from Fukushima
- Section VII. Resolution Plan for Recommendation 11.3, Efficacy of Real Time Radiation Monitoring in Emergency Planning Zone and Onsite
- Section VIII. Resolution Plan for Recommendation 11.4, Training in the Local Community on Radiation, Radiation Safety, and the Use of Potassium Iodide

The staff intends to resolve these recommendations, with the exception of Recommendations 9.3 (partial), 10.3c, 11.2, and 11.4, in a SECY paper to be provided in approximately one year. Generally, this is either to allow time for further documentation or stakeholder interactions as part of the process for closing these items. As discussed in this enclosure, subject to Commission approval, the staff plans to close Recommendations 9.3 (partial), 10.3c, 11.2, and 11.4 based on the progress made to date.

**Section I. Recommendations Subsumed into the Mitigation of Beyond-Design-Basis Events Rulemaking**

As approved in staff requirements memorandum (SRM) to SECY-14-0046, "Fifth 6-Month Status Update on Response to Lessons-Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 9, 2014 (ADAMS Accession No. ML14218A703), several of the Tier 3 EP items are being addressed through the Mitigation of Beyond-Design-Basis Events (MBDBE) rulemaking. These recommendations are as follows:

- Recommendation 9.1 to initiate rulemaking to require EP enhancements for multiunit events;
- Recommendation 9.2 to initiate rulemaking to require EP enhancements for prolonged SBO;
- Recommendation 9.3 (partial) to order licensees to perform various EP enhancements until the rulemaking is complete;
- Recommendation 10.2 to improve command and control structures; and
- Recommendation 11.1 to enhance resources to bring response equipment on site.

Since these are being addressed through the MBDBE rulemaking, they are no longer being tracked separately. In addition to these recommendations, the rulemaking also codifies Orders EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012 (ADAMS Accession No. ML12054A735), and EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," dated March 12, 2012, along with other safety enhancements. The Commission approved the publication of the draft rule for comment, subject to certain changes, in SRM-SECY-15-0065, "Proposed Rule: Mitigation of Beyond-Design-Basis Events (RIN 3150-AJ49)," dated April 30, 2015 (ADAMS Accession No. ML15239A767). The final rule is expected to be submitted to the Commission by December 2016.

The staff has concluded that Recommendation 10.1, to analyze current protective equipment requirements for emergency responders and guidance based upon insights from the accident at Fukushima, is also being addressed by the MBDBE rulemaking. The staff has developed a Draft Regulatory Guide, DG-1319, "Integrated Response Capabilities for Beyond-Design-Basis Events," which will be published with the proposed rule and would endorse the Nuclear Energy Institute (NEI) guidance document NEI 13-06, "Enhancements to Emergency Response Capabilities for Beyond-Design-Basis Events and Severe Accidents," issued September 2014 (ADAMS Accession No. ML14269A230). NEI-13-06 contains the following statement:

> For multiunit sites, ensure that sufficient quantities of radiation protection equipment and supplies are, or can be made available to support protracted operation of an expanded Emergency Response Organization (ERO).

Because Recommendation 10.1 is being addressed by the MBDBE rulemaking, the staff is no longer tracking this recommendation as a separate item.

## Section II. Resolution Plan for Recommendation 9.3 (Partial)

### Background

NTTF Recommendation 9.3 included a recommendation that the NRC order licensees to maintain ERDS capability throughout an accident until the proposed MBDBE rulemaking is complete. When the ERDS voluntary participation program was introduced in August 1989 (Generic Letter (GL) 89-15, "Emergency Response Data System"), the NRC indicated that the implementation of the program was intended to have minimal impact on licensees. To this end, data points to be collected by ERDS were limited and licensees were not required to install new sensors if they did not already collect the requested data. The NRC staff also made it clear that ERDS was not considered to be a safety system.

The following is an excerpt from GL 89-15:

> Will the ERDS be considered safety grade or require redundant equipment?
>
> No. The ERDS feed will be as reliable as the current licensee equipment providing data to the licensee's own TSC [Technical Support Center] and EOF [Emergency Operations Facility]. The addition of new plant instrumentation or computer data points to provide ERDS data will not be required.

### Current Status

Because of work on other, higher-priority NTTF recommendations, the NRC staff has not made progress on this recommendation since the development of the initial project plans in SECY-12-0095.

### Discussion

The enhancements needed to address this recommendation would result in considerable financial burden upon licensees and, in some cases, may not be technically feasible. Since ERDS was not intended to be a safety system, enhancements to provide for redundancy, emergency power, cyber security, or other performance attributes would likely be challenging. While rigorous cost/benefit and technical feasibility analyses have not been performed, the staff believes that these analyses would ultimately demonstrate that it would be impractical and cost prohibitive to proceed.

### Stakeholder Interactions

The NRC staff provided the Fukushima subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015. There has not been any other significant internal or external stakeholder interaction regarding this issue.

**Conclusion**

Subject to Commission approval and based on the staff's assessment provided above, the staff is closing Recommendation 9.3.

## Section III. Resolution Plan for Recommendation 10.3a, Alternative Method for Transmitting ERDS

### Background

NTTF Recommendation 10.3 includes a recommendation that the NRC evaluate ERDS to determine an alternate method (e.g., via satellite) to transmit ERDS data that does not rely on hardwired infrastructure that could be unavailable during a severe natural disaster.

On August 29, 2005, Hurricane Katrina made landfall on the Gulf Coast of the U.S., adversely affecting communities, critical infrastructure, and communications systems. Hurricane Katrina directly affected three nuclear power plants regulated by the NRC: Grand Gulf Nuclear Station, River Bend Station, and Waterford Steam Electric Station. Although there were no safety impacts to these plants, their operational status was affected due to the instability of the regional electrical grid and loss of landline communications. As a result, communications were handled by backup satellite equipment, which was used to maintain contact between the NRC's regional offices, State responders, and affected nuclear power plants. In the case of Waterford, the NRC found that although portable satellite communications equipment was available, it did not function reliably, due to heavy cloud cover and operators were required to be outside for it to work. Further, cellular telephones were not functioning reliably due to infrastructure damage. It was discovered that some phones in "radio mode" were functioning, along with text messaging and email.

As a result of its assessment of these issues, the Commission noted in a March 30, 2006, report entitled, "Task Force Report - 2005 Hurricane Season Lessons Learned Final Report" (ADAMS Accession No. ML060900005) that the NRC should assess agency communications equipment and services associated with emergency notifications systems and recommend improvements in diversity and reliability as a high-priority activity.

This recommendation is consistent with previous recommendations and lessons learned experienced during similar events, such as those that occurred during Hurricanes Andrew (1992) and Gloria (1985). For example, the Hurricane Andrew task force recommended that portable satellite communications equipment be made available in the event of a loss of landline communications.

Following the events of Hurricane Katrina, there have been recent events such as Hurricanes Rita (2005) and Wilma (2005), and the earthquakes in Fukushima, Japan (2011) and Mineral, Virginia (2011), all directly impacting communications systems at commercial nuclear power plants. The adverse impacts of these events heightened the need for the NRC to assess the need for an alternative transmission solution (ATS) for ERDS to improve the reliability of data transmission. Subsequently, an ERDS alternative network study was planned for in 2010, and the work was included as an optional task in the ERDS operations and maintenance contract that was awarded the same year. The task was executed in 2013 and Project Performance Corporation conducted the study.

The study looked into potential options capable of satisfying the network alternative requirements. The proposed solution is a multipoint redundant backup meshed data network

based on cellular and satellite communication systems to lessen the impact of terrestrial outages whenever possible.

**Current Status**

The completed study serves only as a proof of concept and further assessment is needed to determine if this solution should be pursued. The staff is considering addressing this recommendation in two phases. First, a pilot project will be conducted in close collaboration with a selected plant as a “test site” of the proposed strategy. Second, based on the successful completion of the pilot project, the solution would be fully implemented with all nuclear power plants.

The proposed initiative would require a collaborative effort between the NRC and licensee stakeholders to define requirements, identify possible alternatives, and develop a solution design. The identification and design of the alternative transmission solution would not only take into consideration the needs of ERDS, but other NRC program uses that can leverage this capability as well.

**Discussion**

The primary driver for the study/initiative discussed above was primarily the recommendations in the NRC’s March 30, 2006, report. Because the NRC and licensees have been focused on higher-priority post-Fukushima initiatives, significant progress has not been made in assessing the feasibility of pursuing the proposed solution.

Implementation of the proposed initiative would require a collaborative effort between the NRC and licensee stakeholders to ensure that all requirements for developing a design solution are identified. The identification and design of the ATS would not only take into consideration the needs of ERDS, but other NRC program uses that can leverage this capability as well. A detailed analysis will also need to be conducted to choose appropriate cellular and satellite communication vendors and ensure that physical and cyber security requirements are met.

While the recommendation to “determine an alternate method (e.g., via satellite) to transmit ERDS data that does not rely on hardwired infrastructure that could be unavailable during a severe natural disaster” has been addressed, the staff recognizes that ERDS is not a safety system, and as such working on the implementation of the ATS at this time may not be a priority task. With this understanding, the staff currently plans to form an internal working group to review the results of the ATS. The staff recommendations would then be provided in the SECY paper in approximately one year. This item might possibly be considered in the future as part of routine NRC assessments of emergency response capabilities, based on the availability of funds.

**Stakeholder Interactions**

The NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff’s plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on

November 5, 2015. There has not been any other significant internal or external stakeholder interaction regarding this issue.

Additional stakeholder interactions may be necessary as the staff evaluates this issue more fully, depending on the staff's proposed closeout approach.

**Conclusion**

As discussed above, the staff intends to resolve this recommendation in a SECY paper to be provided in approximately one year. This will allow time for further assessment and documentation, along with internal and external engagement, as part of the process for resolving this recommendation.

## Section IV. Resolution Plan for Recommendation 10.3b, ERDS Data Set

### Background

NTTF Recommendation 10.3 included a recommendation that the NRC evaluate ERDS to determine whether the data set currently being received from each site is sufficient for modern assessment needs.

According to NUREG-1394, Revision 1, "Emergency Data System (ERDS) Implementation," issued June 1991 (ADAMS Accession No. ML080790038):

> The tests of the ERDS concept have demonstrated that there is great value in using electronic data transmission for obtaining a limited set of reliable, time tagged data. The NRC response teams functioned more efficiently and their assessments were timelier. Major improvements in ability to focus on the significant factors and to predict the course of events were noted. The questions that were asked of the licensee were focused on overall status and course of action rather than simple data requests, therefore reducing the volume of communication and increasing the quality of the communication.

According to the Statements of Consideration for the final ERDS rule (50 *Federal Register* (FR) 40178, August 13, 1991):

> Although the ERDS data does not portray every detail of a nuclear power reactor in an emergency situation, the Commission believes it does provide the data required by the NRC to perform its role during an emergency. The ERDS parameter list was selected based on the information the NRC Technical Teams need to perform their emergency response functions. Moreover, the set of ERDS data will not be the only input to the NRC. The Emergency Notification System (ENS), a voice communication system, will still be available to transmit data and any other relevant information that is not available through ERDS. In combination, the NRC will receive the necessary information to develop timely and appropriate evaluations of the event and to develop the necessary support actions to ensure protection to public health and safety.

As part of exercise feedback, members of the NRC's Reactor Safety Team and incident responders have expressed a need to review the data set currently being received from each site and determine whether the data are sufficient for future assessment needs.

### Current Status/Discussion

Because of work on other, higher-priority NTTF recommendations, the NRC staff has not made progress on this recommendation since the development of the initial project plans in SECY-12-0095.

**Stakeholder Interactions**

The NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015. There has not been any other significant internal or external stakeholder interaction regarding this issue.

As discussed above, internal stakeholder interactions among Reactor Safety Team members and other incident responders will be needed to determine whether a change to the current ERDS data set requirements is needed. External outreach to stakeholders will also be necessary to determine whether licensees collect any other data points not currently provided to the NRC.

**Conclusion**

The staff intends to resolve this recommendation in a SECY paper to be provided in approximately one year. This will allow time for further assessment and documentation, along with internal and external engagement, as part of the process for resolving this recommendation.

## Section V. Resolution Plan for Recommendation 10.3c, ERDS Continuous Transmission

### Background

Among the items under NTTF Recommendation 10.3, the third item recommended that the NRC evaluate ERDS to determine whether it should be required to transmit continuously, such that no operator action is needed during an emergency.

According to NUREG-1394, Revision 1 (ADAMS Accession No. ML080790038):

> The ERDS would be for use only during emergencies and would be activated by the licensees during declared emergencies classified at the ALERT or higher level to begin transmission to the NRC Operations Center.

According to the Statements of Consideration for the final ERDS rule (50 FR 40178, August 13, 1991):

> The ERDS is designed to transfer needed reactor data from a nuclear power plant only during emergencies. It is not a system to constantly monitor any licensee. The concept of constant monitoring, such as the Nuclear Data Link, was considered after the Three Mile Island accident in 1979, but after much evaluation and deliberation, Congress did not approve the concept for funding.

### Current Status

Currently, there are seven licensees (40 units at 26 sites) that voluntarily transmit ERDS data continuously.

### Discussion

The NRC's approach to date has been to encourage licensees to voluntarily transmit ERDS data sets continuously by highlighting the advantages to them. For example, nuclear power plants that have opted to transmit ERDS data continuously do not have to conduct quarterly tests, as no input will be needed from their operators to support the testing. These licensees are notified of upcoming tests dates, completion, and results. These licensees would also not be challenged to meet the requirement to activate ERDS within 60 minutes of the onset of an event.

Requiring licensees to continuously transmit ERDS data would necessitate rulemaking to change the existing requirements. However, the staff finds it unlikely that such a rulemaking would provide a substantial cost-justified safety improvement, as needed to impose additional requirements on licensees pursuant to 10 CFR Part 50, Section 50.109, "Backfitting." The staff's assessment is that the level of effort and resources needed for a rulemaking would far outweigh the benefits. The staff believes that the NRC should continue to use different forums with licensees to highlight the advantages and encourage them to adopt continuous transmission voluntarily. Therefore, the staff's assessment is that this recommendation should be closed.

**Stakeholder Interactions**

The NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015. There has not been any other significant internal or external stakeholder interaction specific to this recommendation.

However, as discussed above, the staff has and will continue to interact with external stakeholders to highlight the advantages of transmitting ERDS data continuously and encourage them to adopt continuous transmission voluntarily.

**Conclusion**

Subject to Commission approval, and based on the staff's assessment provided above, the staff is closing Recommendation 10.3c.

## Section VI. Resolution Plan for Recommendation 11.2, Evaluate Recovery and Reentry Insights from Fukushima

### Background

NTTF Recommendation 11.2 recommended that the NRC work with the Federal Emergency Management Agency (FEMA), States, and other external stakeholders to evaluate insights from the implementation of EP at Fukushima to identify potential enhancements to the U.S. decision making framework, including the concepts of recovery and reentry.

Specific guidance for response to a nuclear or radiological event is contained in the current version of the Nuclear/Radiological Incident Annex (NRIA) to the National Response Framework, which can be found at https://www.s.gov/media-library/assets/documents/25554. The NRIA is being revised to address recovery in addition to response and will include lessons learned from the Fukushima event as well as other evaluations, such as those related to radiological dispersal devises. FEMA is leading an interagency effort to make those revisions.

The revised NRIA will be an annex to both the Response and Recovery Federal Interagency Operational Plans. It references the needs of State and local response organizations as they relate to relocation and reentry of those displaced by a radiological event.

### Current Status

The NRC staff continues to work with other Federal agencies on revising the NRIA. The draft revised NRIA was used during the first phase of the Southern Exposure 2015 exercise, which was conducted in July. All involved Federal departments and agencies reviewed the draft, and the revised draft was used during the Southern Exposure recovery tabletop exercise in September 2015. Lessons learned during all phases of the Southern Exposure exercise will be incorporated into the document prior to its final release, which is currently planned for later in 2015.

### Discussion

The NRC staff is continuing to work as part of its normal interagency activities related to updates and improvements to the NRIA. These interactions have included consideration of lessons learned from the Fukushima event, as well as other analyses and the insights from various exercises and actual emergencies. These efforts will continue to take advantage of the experiences in Japan with the cleanup and approved reentry of populations to evacuated areas, such as the September 2015 evacuation order that was lifted for the Japanese town of Naraha, about 12 miles south of the Fukushima Dai-ichi nuclear plant site.

Because of progress made to date in addressing this recommendation, the NRC staff's assessment is that Recommendation 11.2 should be closed. Additional interagency collaboration on NRIA updates will be conducted consistent with normal agencies' processes and practices. The NRC will continue to keep the Commission informed of developments in this area, as appropriate.

**Stakeholder Interactions**

The NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015.

As discussed above, the staff has had extensive engagement with other Federal agencies as part of updating the NRIA. The core planning team for this revision includes the U.S. Department of Defense, the U.S. Department of Energy, the U.S. Department of Homeland Security, the Environmental Protection Agency, the Federal Bureau of Investigations, FEMA, and the NRC. Interactions with these stakeholders are expected to continue as the NRIA update is completed.

**Conclusion**

Subject to Commission approval and based on the staff's assessment provided above, the staff is closing Recommendation 11.2.

## Section VII. Resolution Plan for Recommendation 11.3, Efficacy of Real-Time Radiation Monitoring in Emergency Planning Zone and Onsite

### Background

NTTF Recommendation 11.3 recommended that the NRC staff study the efficacy of real-time radiation monitoring on site and within the emergency planning zones (EPZs) (including consideration of alternating current power independence and real-time availability on the internet). There is extensive regulatory history associated with the evaluation of real-time radiation monitoring on site and within the EPZs, in which policy decisions have been previously made regarding their efficacy. In its evaluation, the NTTF concluded that "as long as field teams are adequately staffed, equipped, and capable of transit given the nature of the natural disaster, field monitoring remains an effective method to acquire radiation data."

### Current Status

In response to the recommendation, the staff is documenting the historical background of real time radiation monitoring systems and the justifications for the historical decisions. In December 1980, the staff included a requirement for "radiation exposure meters (continuous indication at fixed locations)," in addition to requirements for "plant and environs radiation (portable instrumentation)," in Regulatory Guide (RG) 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," revised June 2013. However, in the March 1983 revision of the guide, staff omitted the requirement for continuous indication, noting that,

> It is unlikely that a few fixed-station area monitors could provide sufficiently reliable information to be of use in detecting releases from unmonitored containment release points. However, there may be circumstances in which such a system of monitors may be useful. The decision to install such a system is left to the licensee.

Based upon insights from the severe accident studies conducted since the TMI accident, the staff has taken the position that the initial protective action recommendation made by the licensee upon declaration of a General Emergency be based primarily on plant conditions rather than radiological assessment data. This protocol has the capability to initiate protective actions before the onset of the release. The Japanese protocol at the time of the Fukushima accident was to base protective action recommendations upon actual radiation measurements. Wind direction in the early days of the event was toward offshore and away from the fixed radiation monitors. There is also anecdotal information that the fixed area radiation monitors surrounding Fukushima were adversely affected by the tsunami.

However, because of work on other, higher-priority NTTF recommendations, the NRC staff has not made substantive progress on this recommendation since the development of the initial project plans in SECY-12-0095, and additional assessment and documentation is needed.

### Discussion

The NRC staff is investigating the basis for the inclusion of this guidance in Revision 2 of RG 1.97 and its removal in Revision 3. The staff will evaluate the lessons learned from

Fukushima with regard to the radiation monitoring systems in use at the time of the event to discern whether this experience warrants reconsideration of the actions taken with RG 1.97, Revision 3, and subsequent revisions. However, the staff finds it unlikely that changes to require real-time radiation monitoring would provide a cost-justified substantial safety improvement, as needed to impose additional requirements on licensees under 10 CFR 50.109.

**Stakeholder Interactions**

The NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015. There has not been any other significant internal or external stakeholder interaction specific to this recommendation.

**Conclusion**

As discussed above, the staff intends to resolve this recommendation in a SECY paper to be provided in approximately one year. This will allow time for further assessment and documentation, along with internal and external engagement. The followup SECY will include a more detailed discussion of historical assessments and their applicability to the closure of this recommendation.

## Section VIII. Resolution Plan for Recommendation 11.4, Training in the Local Community on Radiation, Radiation Safety, and the Use of Potassium Iodide

### Background

NTTF Recommendation 11.4 recommended that the NRC conduct training, in coordination with the appropriate Federal partners, on radiation, radiation safety, and appropriate use of potassium iodide (KI) in the local community around each nuclear power plant.

The requirements related to public information and education are contained in 10 CFR 50.47(b)(7) and (15), as well as Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities" (Section IV.D.2) to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," issued November 1980, provides guidance to licensees and State and local authorities on meeting those standards.

Additional guidance from FEMA to State, local, and Tribal authorities on the development of radiological emergency plans is contained in FEMA's Radiological Emergency Preparedness (REP) Program Manual. This manual provides supplemental guidance on KI for the public located within the plume EPZ.

As discussed below, in accordance with a FEMA/NRC Memorandum of Understanding (MOU), FEMA takes the lead in developing public information and educational programs.

### Current Status

The adequacy of State, local, and Tribal REP plans and preparedness is continuously evaluated by FEMA based on: (1) the review and approval of significant changes to offsite REP plans; (2) the annual letter of certification (including a detailed review of the public education and information process) provided by the respective State emergency management agency to FEMA; and (3) evaluation by FEMA of the biennial REP exercises. The FEMA REP Program Manual also provides guidance for review and evaluation of public information materials distributed by offsite response organizations and licensees for nuclear power plants, as well as a comprehensive public information review checklist.

FEMA has extensive guidance on public information and education for the community around each nuclear power plant, including radiation, radiation safety and the appropriate use of KI, which is periodically assessed to ensure continued reasonable assurance process. According to FEMA, several State and local emergency management authorities have looked at the adequacy of their public outreach materials related to a commercial nuclear power plant event, and have been or are revisiting them subsequent to the Fukushima Dai-ichi event.

### Discussion

Based on the FEMA/NRC MOU, contained in Appendix A to 44 CFR Part 353, Section II.H (Public Information and Education Programs), FEMA will take the lead in developing public

information and educational programs. The NRC will assist FEMA by reviewing, for accuracy, educational materials concerning radiation, and its hazards and information regarding appropriate actions to be taken by the general public in the event of an accident involving radioactive materials.

FEMA is engaging respective Federal, State, and local REP expertise, as well as developing a partnership with industry, as part of the activities for an REP Program Outreach Integrated Process Team. The REP Program Outreach Integrated Process Team will engage stakeholders through forums, such as the Federal Radiological Preparedness Coordinating Committee, Conference of Radiation Control Program Directors, National REP Conference, National Emergency Management Association, and the International Association of Emergency Managers.

Because of progress made to date by FEMA in engaging the public around nuclear plant sites and using lessons learned and sharing best practices, the NRC staff's assessment is that Recommendation 11.4 should be closed. Additional collaboration between NRC and FEMA on related activities will continue as part of the FEMA/NRC MOU. The NRC staff will continue to monitor and work with FEMA as part of the FEMA/NRC Steering Committee on Emergency Planning, established under the interagency MOU, and will provide information with respect to training awareness to the Commission, as part of the annual SECY paper on emergency preparedness and incident response.

**Stakeholder Interactions**

The NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015.

As discussed above, the NRC has and will continue to engage with FEMA as part of the activities related to the NRC/FEMA MOU.

**Conclusion**

Subject to Commission approval and based on the staff's assessment provided above, the staff is closing Recommendation 11.4.

**Enclosure 8: Proposed Resolution Plan for Tier 3 Recommendation 12.1 - Enhancements to the Reactor Oversight Process**

**Background**

Near-Term Task Force Recommendation 12.1 recommended that the U.S. Nuclear Regulatory Commission (NRC) expand the scope of the annual Reactor Oversight Process (ROP) self-assessment and biennial ROP realignment to more fully include defense-in-depth considerations. In SECY-11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," dated October 5, 2011 (Agencywide Document Access and Management System (ADAMS) Accession No. ML11272A111), Recommendation 12.1 was prioritized as a Tier 3 activity because of its dependency on the resolution of Recommendation 1 to establish a regulatory framework that balances defense-in-depth and risk considerations. The initial project plan for Recommendation 12.1 in SECY-12-0095, "Tier 3 Program Plans and 6 Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012 (ADAMS Accession No. ML12165A092), deferred work on Recommendation 12.1 until Commission direction was received on Recommendation 1.

In the staff requirements memorandum to SECY-13-0132, "U.S. Nuclear Regulatory Commission Staff Recommendation for the Disposition of Recommendation 1 of the Near-Term Task Force Report," dated May 19, 2014 (ADAMS Accession No. ML14139A104), the Commission directed the staff to close Recommendation 1 and to reevaluate defense-in-depth in the context of the Commission direction on a longer-term Risk Management Regulatory Framework (RMRF) project. When completed, the RMRF should provide options to the Commission for adopting a more comprehensive, risk-informed, performance-based regulatory approach to ensure the continued safe and secure use of nuclear material. The work on RMRF is still ongoing, will depend on future Commission direction, and is expected to take many years to fully complete.

**Current Status**

Besides the ongoing work on RMRF, the staff has been, and will continue, using existing agency processes to enhance the ROP. As part of the Baseline Inspection Procedure Enhancement project, the NRC staff is working to identify and evaluate improvements to the ROP based on insights from Fukushima-related lessons-learned, reviews, and inspection activities. For example, the NRC staff has identified and implemented improvements to the ROP inspection program from post-Fukushima inspections of licensee walkdowns of flood protection features. Specifically, in December 2014, the NRC staff completed proposed changes to Inspection Procedure (IP) 71111.01, "Adverse Weather Protection," to incorporate lessons learned from these walkdowns. The revised procedure has been approved and will be available for the upcoming inspection cycle.

In 2014, the NRC staff issued Temporary Instruction (TI)-2515/191, "Inspection of the Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communication/Staffing/Multi-Unit Dose Assessment Plans," dated October 6, 2014, to verify licensee's compliance with Order EA-12-049, "Issuance of Order to

Modify Licenses with Regard to Requirements for Mitigation Strategies For Beyond-Design-Basis External Events," dated March 12, 2012 (ADAMS Accession No. ML12054A735), and other Tier 1 items such as Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," dated March 12, 2012 (ADAMS Accession No. ML12056A044), and implementation of enhancements to emergency preparedness staffing and communication plans and multiunit dose assessment capabilities completed in response to Recommendation 9.3. The first compliance inspection associated with these two orders and Recommendation 9.3 was completed in March 2015 at Watts Bar Nuclear Plant, Unit 1. Feedback from the pilot inspection has been collected, is under evaluation, and will be used to enhance TI-2515/191. Additional TI-2515/191 inspections will be completed as reactor sites come into compliance with Orders EA-12-049 and EA-12-051. The staff plans to periodically evaluate the information gathered from implementing TI-2515/191 to update the temporary instruction procedure using Inspection Manual Chapter (IMC) 0040, "Preparing, Revising, and Issuing Documents for the NRC Inspection Manual."

Using existing processes, the staff will assess best practices for documenting and evaluating inspection feedback from TI-2515/191 and other Fukushima lessons-learned activities. IMC 0801, "Reactor Oversight Process Feedback Program," will be used to collect feedback from the TI-2515/191 inspections. The ROP Feedback Program is used by the staff to identify issues that need program-level attention, and to suggest changes to improve the effectiveness of the ROP. Besides the work on TI-2515/191 and the enhancements to IP-71111.01, information obtained from other Fukushima lessons-learned activities is also being assessed commensurate with NRC prioritization, rulemaking, and licensee implementation schedules.

In addition, the NRC staff is revising both the ROP self-assessment and realignment programs. The staff is evaluating insights to improve ROP's effectiveness and scope. These enhancements are being pursued outside of the Fukushima-related lessons-learned activities. Under the revised ROP self-assessment process, the staff will make focused evaluations of specific areas of the program and perform effectiveness reviews of recent changes to the program to ensure the intended improvements were realized. The inclusion of defense-in-depth considerations could be selected as a focus area in a future ROP self-assessment, depending on the outcome of RMRF or other initiatives.

**Discussion**

Dependency on the RMRF

The staff is currently developing a SECY paper with a proposed path forward on the RMRF to obtain Commission direction on the use of defense-in-depth in the NRC's regulatory program. The SECY paper on RMRF is due to the Commission by December 2015. However, insights on the use of defense-in-depth based on Commission direction on that paper may not be available in time to support the resolution of Recommendation 12.1. For this reason, the staff proposes that the resolution of Recommendation 12.1 no longer be considered dependent on the RMRF project. Instead, the staff is proposing that the resolution be based on potential ROP improvements coming from Fukushima-related inspection activities, such as feedback gathered from TI-2515/191, and the work being done to revise the ROP outside the Fukushima lesson-

learned activities. Any ROP enhancements necessary as a result of Commission direction on RMRF would also be addressed outside the scope of Fukushima lessons-learned activities.

Path Forward on ROP Enhancements

As discussed above, work is ongoing to obtain insights from the completed and planned TI-2515/191 inspections and to incorporate these insights into the ROP. These inspections are being scheduled based on licensees' compliance dates for the associated requirements, with the majority of the TI-2515/191 inspections scheduled for completion in 2016 and 2017. In addition to verifying that Orders EA-12-049 and EA-12-051 and activities associated with Recommendation 9.3 are appropriately implemented, these inspections are expected to provide the staff with a better understanding of the inspection and oversight of post-Fukushima regulatory requirements associated with the protection and mitigation of beyond-design-basis events, and as such, will be used to inform longer-term ROP enhancements.

To evaluate potential future ROP enhancements, the staff plans to:

- Use the existing ROP Feedback Program, described in IMC 0801, to incorporate insights from TI-2515/191 inspections into the ROP.

- Ensure that future ROP enhancements from other Fukushima lessons-learned activities, such as seismic and flooding walkdowns and containment vent requirements, are appropriately considered.

- Assess and implement additional ROP enhancements in response to the Fukushima rulemaking activities (e.g., through development of a program to explicitly provide periodic oversight of industry's implementation of the severe accident management guidelines (SAMGs), as directed by the Commission in the staff requirements memorandum to SECY-15-0065 (ADAMS Accession No. ML15239A767)).

- Ensure effective communication with the NRC's regional offices and other stakeholders is maintained on issues of interest, such as the regulatory treatment of reevaluated hazards, maintenance rule impacts of mitigating strategies, and the development of significance determination processes.

Because of the proposed path forward to consider the potential ROP enhancements from the Fukushima-related inspection activities, and because of ongoing efforts outside the Fukushima lessons-learned activities to revise the ROP, the staff proposes that Recommendation 12.1 be closed.

**Stakeholder Interaction**

The NRC staff provided the Fukushima subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015. The staff does not plan to meet with ACRS or

conduct additional stakeholder interactions specific to this recommendation, although such interactions may take place on future ROP enhancements outside the scope of this recommendation, if appropriate.

**Conclusion**

Subject to Commission approval, based on progress made to date in incorporating Fukushima lessons learned into the ROP and the staff's plans to use well-established processes to make any necessary future enhancements, Recommendation 12.1 is closed.

**Enclosure 9: Proposed Resolution Plan for Tier 3 Recommendation 12.2 - Enhancements to Severe Accident Training for NRC Staff and Severe Accident Management Guidelines for Resident Inspectors**

**Background**

Near-Term Task Force Recommendation 12.2 recommended that the U.S. Nuclear Regulatory Commission (NRC) enhance staff training on severe accidents, including resident inspector training on severe accident management guidelines (SAMGs). In SECY-11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11272A111), Recommendation 12.2 was prioritized as a Tier 3 activity because it was dependent on the resolution of Recommendation 8, now part of the Mitigation for Beyond-Design-Basis Events (MBDBE) rulemaking. The initial project plan in SECY-12-0095, "Tier 3 Program Plans and 6 Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated June 13, 2012 (ADAMS Accession No. ML12165A092), stated that the staff would assess the current level of NRC staff training conducted on severe accidents and SAMGs and consider future training enhancements.

As part of its initial plan, the NRC staff proposed both near- and long-term action items. The near-term activities included the following:

- Review the frequency of existing severe accident training courses.
- Update existing severe accident training to include Fukushima lessons learned.
- Revise NRC qualification program training requirements for severe accidents.
- Meet with stakeholders to more fully inform training enhancements.

The long-term enhancements, initially planned to be done by 2016, included the following:

- Enhance training based on the State-of-the-Art Reactor Consequence Analysis (SOARCA) study and followup research activities, the Level 3 probabilistic risk assessment (PRA) study, and longer-term Fukushima lessons learned initiatives.
- Revise NRC staff qualification training to include training on the Fukushima accident and SAMGs.
- Develop new training courses on severe accident progression, consequences, and SAMGs.
- Meet with stakeholders to more fully inform training enhancements.

**Current Status**

Significant progress has been made in addressing this recommendation and additional training enhancements are planned. Current progress and planned activities associated with these items (both short and long-term enhancements) are summarized below:

- Review the frequency of existing severe accident training courses: The NRC staff has evaluated the frequency of the different training courses associated with severe accidents in nuclear power plants. Such courses include PRA and reactor technology courses available for NRC staff. Given the demand for attendance and availability of resources, the current frequency of severe accident related courses was found to be adequate. These courses are often part of reactor operation and technical qualification programs that provide a further perspective on the subject matter and exposure to additional courses updated to include information on the Fukushima accident.

- Update existing NRC staff severe accident training to include Fukushima lessons learned: The NRC staff has generated and gathered significant information about the Fukushima accident and lessons learned. This information has been used to update existing severe accident training (e.g., R-800, "Perspectives on Reactor Safety") and to develop new courses to include lessons learned from the Fukushima accident, as described below.

- Revise NRC qualification program training requirements for severe accidents: The NRC staff reviewed existing qualification programs with requirements for training on severe accidents. The review found that some qualification programs should be improved to include study activities associated with the Fukushima accident. For example, in the case of NRC's Office of Nuclear Regulation (NRR) qualification program, the staff completed an enhancement to expand individual study activities associated with severe accidents to include references with information about the Fukushima accident and actions taken by the NRC staff to address the lessons learned. The staff is planning future revisions to other qualification programs, such as those used for reactor inspectors, to incorporate training on SAMGs.

- Enhance training based on the SOARCA study and followup research activities: Knowledge from SOARCA and followup activities has been used to enhance and develop new training courses on severe accident progression. Technical staff in this area will consider ongoing and future research work in SOARCA, and any future Fukushima lessons-learned insights that could generate additional training enhancements.

- Develop new training courses on severe accident progression, consequences, and SAMGs, as determined to be necessary: The NRC staff has been conducting numerous agency wide seminars on the state-of-the-art understanding of severe accidents. To date, seven different severe accident progression seminars have been conducted and made available to NRC staff. Seminar topics have included severe accident-induced steam-generator tube rupture; in-vessel melt progression and retention; steam

explosions; direct containment heating; fission product release, transport, and deposition; and a comparison of the Fukushima accident sequence with a state-of-the-art analysis model. These seminars are generally led by at least two experts (an in-house expert and an external expert) to offer diverse perspectives that may enable the NRC staff to better understand each severe accident phenomenon. The seminars began in March 2014 and are being conducted quarterly. Each future seminar will cover one severe-accident phenomenon (e.g., hydrogen combustion). Video recordings of the seminars have and will continue to be made available to NRC staff in iLearn for knowledge management purposes.

The NRC staff is currently reviewing options to incorporate SAMG training into the qualification program for NRC resident inspectors. When complete, this activity will better enable inspectors to oversee the implementation of SAMGs by licensees in the very unlikely event of a severe accident. Other relevant subjects may be added to the course curriculum. The course should be available to the NRC staff by the end of Fiscal Year 2016.

- Revise NRC qualification training to include training on SAMGs: As discussed above, the NRC staff is working to incorporate SAMG training in the appropriate qualification programs. Once the training is available, the staff plans to revise the NRC's inspector qualification program to incorporate training on SAMGs.

- Inform stakeholders of new training enhancements: Once the new SAMG training course is available, the staff will inform internal stakeholders, including the NRC's regional offices, of all training enhancements to ensure staff is aware of these tools.

**Discussion**

The NRC staff has obtained substantial information on severe accidents, both through lessons learned from the Fukushima accident, research related to the accident, and previous studies. This information has been and will continue to be made available to the NRC staff via new and updated training tools. Research activities on severe accidents will continue to enhance understanding of severe accident phenomenology and will lead to additional insights and future knowledge management opportunities.

Based on completed and ongoing activities, the staff has achieved the objectives of Recommendation 12.2. Remaining activities include verifying that the new SAMG training course for the NRC staff is made available, updating the inspector qualification program to incorporate training on SAMGs, and generating a written communication with all training enhancements to share with the NRC staff. Because of progress already made on this recommendation and the fact that the remaining activities are well-underway using established NRC processes, the staff considers Recommendation 12.2 to be closed.

**Stakeholder Interaction**

As explained previously, NRC internal stakeholders will be notified of all training enhancements once SAMG training for resident inspectors is available.

The NRC staff provided the Fukushima subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015. The staff does not plan to meet with ACRS or conduct additional stakeholder interactions specific to this recommendation.

**Conclusion**

Subject to Commission approval, based on progress made to date in enhancing training on severe accidents and SAMGs and the staff's plans to used well-established processes to make any necessary future enhancements, Recommendation 12.2 is closed.