Program Plans for Tier 3 Recommendations

Enclosure 3

Contents

Tier 3 – Near-Term Task Force Recommendation 2.2

Program Plan for Periodic Confirmation of Seismic and Flooding Hazards

Purpose

The purpose of this program plan is to define the initial pre-rulemaking activities necessary to position the agency for a future rulemaking to implement Near Term Task Force (NTTF) Recommendation 2.2. As the staff gains experience from the implementation of Recommendation 2.1 and knowledge from the pre-rulemaking activities, the staff will develop a complete rulemaking plan for Recommendation 2.2.

NTTF Recommendation and Other Direction

The Task Force recommended that the U.S. Nuclear Regulatory Commission (NRC) take the following action: "Initiate rulemaking to require licensees to confirm seismic hazards and flooding hazards every 10 years and address any new and significant information. If necessary, update the design basis for SSCs important to safety to protect against the updated hazards."

In SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, the staff prioritized Recommendation 2.2 as Tier 3 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. Recommendation 2.2 suggests a periodic update of the reevaluated hazards based on any new and significant information since the most recent reevaluation. In the December 15, 2011, staff requirements memorandum (SRM) to SECY-11-0137, the Commission agreed with the Tier 3 prioritization of Recommendation 2.2.

Section 402 of the Consolidated Appropriations Act, 2012, (Public Law 112-074, dated December 23, 2011), requires a reevaluation of licensees' design basis for external hazards and expands the scope to include other external events, as described below:

> The Nuclear Regulatory Commission shall require reactor licensees to 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.

The staff interprets this language to indicate that other external hazards, such as those caused by meteorological effects, should be included in the periodic updates that will be required once Recommendation 2.2 is implemented. Recommendation 2.1 for other natural external hazards was identified as a Tier 2 activity because of the lack of availability of the critical skill sets for both the NRC staff and external stakeholders.

For the purposes of Recommendation 2.2, the staff includes seismic, flooding, and other natural and man-related external hazards[1] within the scope of this rulemaking.

Regulations and Guidance

- General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

- GDC 4, "Environmental and Dynamic Effects Design Bases," of Appendix A to 10 CFR Part 50 requires, in part, that SSCs that are important to safety be adequately protected against the effects of missiles resulting from events and conditions outside the plant.

- GDC 44, "Cooling Water," of Appendix A to 10 CFR Part 50 requires, in part, that a system to transfer heat from SSCs important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these SSCs under normal operating and accident conditions.

- 10 CFR Part 100, "Reactor Site Criteria," Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," provides detailed criteria to evaluate the suitability of proposed sites and the suitability of the plant design basis established in consideration of the seismic and geologic characteristics of the proposed sites. Appendix A, which applies to stationary reactor site applications before January 11, 1997, provides a deterministic approach for developing the seismic plant design basis. In contrast, 10 CFR 100.23, "Geologic and Seismic Siting Criteria," which applies to applications dated on or after January 11, 1997, provides an approach for detailed characterization of uncertainties and is being used by new reactor applicants to develop seismic design bases.

- 10 CFR 100.20(b), requires that the nature and proximity of man-related hazards (e.g., airports, dams, transportation routes, military and chemical facilities) be evaluated to establish site parameters for use in determining whether a plant design can accommodate commonly occurring hazards, and whether the risk of other hazards is very low.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," issued November 1975 and updated April 1978, July 1981, and March 2007.

- Regulatory Guide (RG) 1.29, "Seismic Design Classification," issued June 1972 and updated August 1973, February 1976, September 1978, and March 2007.

- RG 1.59, "Design Basis Floods for Nuclear Power Plants," issued August 1973 and updated April 1976 and August 1977.

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[1] Man-related external hazards include military, industrial and transportation facilities.

- RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” issued October 1973 and updated December 1973.
- RG 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” issued April 1974 and updated March 2007.
- RG 1.102, “Flood Protection for Nuclear Power Plants,” issued October 1975 and updated September 1976.
- RG 1.125, “Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants,” issued March 1977 and updated October 1978 and March 2009.
- RG 1.208, “A Performance-Based Approach To Define the Site-Specific Earthquake Ground Motion,” issued March 2007.
- RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” issued October 2011.

Plants that received construction permits before issuance of the GDC in 1971 meet the intent of the GDC. Other regulations that guide the staff’s approach to Recommendation 2.2 include 10 CFR 50.34(a)(1), (a)(3), (a)(4), (b)(1), (b)(2), and (b)(4); 10 CFR 50.109, “Backfitting"; 10 CFR 100.23, “Geological and Seismic Siting Criteria”; Subpart B, “Evaluation Factors for Stationary Power Reactors Site Applications On or After January 10, 1997,” to 10 CFR Part 100; 10 CFR 100.20(c)(2); and 10 CFR 100.21(d).

Staff Assessment and Basis for Prioritization

The staff identified several generic issues in addressing new and significant information related to seismic, flooding, and other external hazards; Recommendation 2.1 subsumes two of these generic issues, Generic Issue-199, “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States,” dated June 9, 2005, and Generic Issue 204, “Flooding of Nuclear Power Plant Sites Following Upstream Dam Failure,” dated February 29, 2012. Furthermore, although there have been several site-specific updates of hazards, no systematic approach has been taken to update the hazards at all sites as new and significant information becomes available.

One of the key issues identified in the implementation of Recommendation 2.2 is how to treat the information gained from the periodic updates of the seismic, flooding, and other external hazards at the site. Depending on what new information licensees obtain during periodic updates, it is possible that changes to the design basis may be needed; however, the staff has not evaluated all possible approaches for the treatment of information gathered as part of these periodic updates. It is possible that, based on the new hazard information, further evaluation may be needed to address any changes the updated hazard may have on the risk assessment. The staff will consider international experience with periodic safety review updates, as well as domestic and international information in determining the best use of the updated information. To fully implement Recommendation 2.2, the staff will need to determine what constitutes “new and significant information” in the context of this periodic update.

The staff does not anticipate that the actions associated with Recommendation 2.2 will have any effect on the completion schedule for Tier 1 Recommendations 2.1 and 2.3 for seismic and flooding hazards, or Tier 2 Recommendation 2.1 for other natural external hazards.

Dependencies

The staff concludes that Recommendation 2.2 is highly dependent upon Tier 1 Recommendation 2.1. As such, the staff's activities, including development of a complete rulemaking plan, will continue to evolve as the staff gains experience from implementation of Recommendation 2.1.

Staff Plan

The staff plans to undertake the following activities:

(1) Begin limited pre-rulemaking activities necessary to position the agency for a future rulemaking to implement NTTF Recommendation 2.2.

(2) Provide a complete rulemaking plan once sufficient experience is gained with implementation of Recommendation 2.1 and the pre-rulemaking activities of Task 1.

Schedule and Milestones

The staff is currently engaged in completing Recommendations 2.1 and 2.3 as they pertain to the reevaluation of seismic and flooding hazards and expects to begin industry interactions related to Recommendation 2.1 for other natural external hazards in calendar year 2013. Because of the complexity of this rulemaking activity and the need to understand how the updated information is used based on international experience with similar updates, the staff plans to use contractor support starting in fiscal year 2013 for pre-rulemaking activities if resources become available. This initial work will focus on collecting and reviewing background information and similar international experiences to support the technical bases for this rulemaking. The staff will also record any decisions made or guidance developed for Recommendation 2.1 that may be applicable to Recommendation 2.2. The staff will determine if an advance notice of proposed rulemaking is necessary given the ongoing interactions with external stakeholders on Recommendation 2.1. The staff will undertake the following activities:

(1) Conduct pre-rulemaking activities, including collection of background information, lessons learned from similar experiences, and insights from Recommendation 2.1. (Minimum of 18 months).

(2) Provide the Commission with updates on pre-rulemaking activities and plans to develop a complete rulemaking plan. (As needed)

Resources

Office	FY 2013		FY 2014		estimated FY 2015 and beyond	
	FTE	Dollars, $K	FTE	Dollars, $K	FTE	Dollars, $K
RES	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
NRR	[ ]	Official Use Only – Sensitive Internal Information				[ ]
NRO	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
TOTAL	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

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Note:

The timing and necessary resources for the activities associated with the initial pre-rulemaking activities for NTTF recommendation 2.2 will not affect the completion of Tier 1 Recommendations 2.1 and 2.3 for seismic and flooding hazards, or Tier 2 Recommendation 2.1 for other external hazards.

Tier 3 – Near-Term Task Force Recommendation 3

Program Plan for Potential Enhancements to the Capability To Prevent or Mitigate Seismically Induced Fires and Floods

Purpose

The purpose of this program plan is to evaluate potential safety enhancements for seismically induced fires and floods and determine whether additional regulatory action (e.g., order, rulemaking) is needed.

NTTF Recommendation and Other Direction

The Task Force recommended that "as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods."

In SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, the staff prioritized Recommendation 3 as Tier 3 because longer term staff evaluation was required to support a decision on the need for regulatory action. In the SRM to SECY-11-0137, the Commission agreed with the Tier 3 prioritization of Recommendation 3, but directed the staff to initiate a probabilistic risk assessment (PRA) methodology to evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods as part of Tier 1 activities. The Commission indicated that the prerequisite activity to initiate development of an appropriate PRA methodology to support this issue should be started without unnecessary delay, while other Recommendation 3 activities remained prioritized as Tier 3.

In SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Tsunami," dated February 17, 2012, the staff summarized pre-planning activities necessary to develop a detailed project plan for the development of a PRA method for seismically induced fires and floods. Pre-planning activities would address several key aspects of this activity including the objectives of the methodology, intended users, stakeholder involvement, coordination with other initiatives, resource needs, and proposed schedule.

Regulations and Guidance

- GDC 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A to 10 CFR Part 50 requires, in part, that 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 3, "Fire Protection," of Appendix A to 10 CFR Part 50 requires, in part, that fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on SSCs important to safety and that fire fighting systems be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of SSCs.

- 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," Section III.O, "Oil Collection System for Reactor Coolant Pump," requires, in part, that the oil collection system be designed, engineered, and installed such that there is reasonable assurance that the system will withstand a safe shutdown earthquake (SSE).

- 10 CFR Part 100, "Reactor Site Criteria," Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," provides detailed criteria to evaluate the suitability of proposed sites and the suitability of the plant design basis established in consideration of the seismic and geologic characteristics of the proposed sites. Appendix A, which applies to stationary reactor site applications before January 11, 1997, provides a deterministic approach for developing the seismic plant design basis. In contrast, 10 CFR 100.23, which applies to applications on or after January 11, 1997 and is being used by new reactor applicants to develop seismic design bases, requires that uncertainties inherent in the estimates of the SSE be addressed through appropriate analysis, such as probabilistic seismic hazard analysis.

- NUREG-0800, Section 2.4.2, "Floods," issued November 1975 and updated June 1978, July 1981, April 1989, and March 2007; Section 2.4.10, "Flooding Protection Requirements," issued November 1975 and updated May 1978, July 1981, and March 2007; and Section 2.5.2, "Vibratory Ground Motion," issued November 1975 and updated July 1981, August 1989, March 1997, and March 2007 provide general review guidance related to site characteristics and site parameters together with site-related design parameters and design characteristics associated with the ground motion response spectrum and flooding hazards.

- RG 1.29, "Seismic Design Classification," issued June 1972 and updated August 1973, February 1976, September 1978, and March 2007, describes one acceptable method for use in identifying and classifying those features of nuclear power plants that must be designed to withstand the effects of the SSE ground motion.

- RG 1.59, "Design Basis Floods for Nuclear Power Plants," issued August 1973 and updated April 1976 and August 1977, discusses the design-basis floods that nuclear power plants should be designed to withstand without loss of capability for cold shutdown and maintenance thereof.

- RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," issued October 1973 and updated December 1973, describes a procedure for defining the response spectra for the seismic design of nuclear power plants.

- RG 1.102, "Flood Protection for Nuclear Power Plants," issued October 1975 and updated September 1976, describes the types of acceptable flood protection for the safety-related SSCs identified in RG 1.29.

- RG 1.125, "Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants," issued March 1977 and updated October 1978 and March 2009, describes the detail and documentation of data and studies that an applicant should include in the preliminary and/or final safety analysis report to support the use of physical hydraulic

model testing for predicting the performance of hydraulic structures and systems for nuclear power plants that are important to safety.

- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," issued February 2004 and updated March 2009, describes one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for LWRs.

- Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," dated February 19, 1987, requested licensees to review the seismic adequacy of certain equipment in operating nuclear power plants against seismic criteria not in use when these plants were licensed.

- GL 88-20, Supplement 4, "Individual Plant Examination Of External Events (IPEEE) for Severe Accident Vulnerabilities," dated June 28, 1991, and Supplement 5, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," September 8, 1995, requested licensees to perform an IPEEE for plant-specific severe accident vulnerabilities initiated by external events and to submit the results to the NRC.

Staff Assessment and Basis for Prioritization

As described in the NTTF Report, seismically induced fires have the potential to cause multiple failures of safety-related SSCs 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 SSCs intended to mitigate the effects of fires and floods. Although these issues have been examined to a limited degree in the Generic Issues Program (e.g., Generic Safety Issue (GSI)-172, "Multiple Systems Responses Program") and responses to GL 88-20, Supplement 5, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," the NTTF concluded that the staff should reevaluate the potential for common-mode failures of plant safety-related SSCs as the result of seismically induced fires and floods.

The staff believes that the capability to prevent seismically induced fires and floods can be enhanced with traditional deterministic design-basis methods. However, the identification of accident sequences and complex dependencies needed to evaluate the mitigation of these events can be done more systematically through PRAs. Consequently, the Commission directed the staff to develop a PRA methodology for seismically induced fires and floods as a necessary prerequisite for the implementation of NTTF Recommendation 3.

As described in SECY-12-0025, the staff has initiated activities to develop a PRA method to address seismically induced fires and floods. There are significant technical challenges associated with this effort including, but not limited to the following:

- hazard definition and characterization;

- quantification of site-specific seismically induced fire ignition frequencies
- quantification of site-specific seismically induced flooding frequencies
- seismic fragilities for SSCs, including fire protection components
- treatment of uncertainties

• modeling concurrent and subsequent initiating events;

• treatment of systems interactions;

• human reliability analysis methodologies suitable for seismically induced hazards, and

• multiunit risk considerations.

No current state-of-practice PRA methods are capable of supporting a quantitative assessment of seismically induced fires and floods for nuclear power plants. Although the RG 1.200 regulatory positions addressing an acceptable approach for defining PRA technical adequacy do consider seismic/fire interactions, only a qualitative assessment is needed. Specifically, RG 1.200 states that a qualitative assessment should be performed to verify that such seismically induced fires have been considered and that steps are taken to ensure that the potential risk contributions are mitigated. Current PRA standards do not specifically address seismically induced fires and floods. However, the American Society of Mechanical Engineers and the American Nuclear Society Joint Committee on Nuclear Risk Management recently formed a working group to investigate approaches for quantitatively addressing multiple concurrent hazards (including seismically induced fires and floods).

<u>Dependencies</u>

The issue of concurrent external hazards is closely related to several other NTTF Tier 1 recommendations associated with external hazard evaluation, seismic risk evaluation, and various plant modifications pertinent to external hazard mitigation capability. Specifically, the NRC issued the following orders and information requests on March 12, 2012:

• EA-12-049, “Issuance of Order To Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events”

• EA-12-050, “Issuance of Order To Modify Licenses with Regard to Reliable Hardened Containment Vents”

• EA-12-051, “Issuance of Order To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation”

• 10 CFR 50.54(f) Letter, “Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident”

Collectively, these activities will improve the characterization of the unique external hazards for each nuclear plant site. However, these orders and information requests are likely to result in changes to the plant licensing basis over the next several years, some of which cannot be fully defined at this

time. Therefore, to gain a more complete understanding of plant-specific hazards, vulnerabilities, mitigation capabilities, and potential post-Fukushima licensing basis changes, it will be necessary to monitor the progress in these Tier 1 areas before substantial resources are dedicated to the evaluation of seismically induced fires and floods.

The timing of some of these Tier 1 activities also depends on the results of initial evaluations and the associated NRC staff evaluation. For example, the Recommendation 2.1 hazard evaluations associated with seismic and flooding events will be implemented using a two-phase multiyear process[2]. During the first phase, licensees will complete a seismic and flooding hazard evaluation and perform a risk evaluation, if necessary. During the second phase, the staff will determine if additional regulatory actions are necessary (e.g., changes to the design and licensing basis) based on the results of the first phase. Based on the summary timeline provided in Table 1, substantially new information from licensees on flooding assessments (e.g., submittal of integrated flooding assessment reports) will not be available until 2015. Similarly, new seismic risk evaluations for high priority plants in the central and eastern United States will not be available until 2016. As stated in SECY-12-0025, the staff anticipates collecting sufficient information to make a regulatory decision for most plants by 2017. Although the Recommendation 2.3 seismic and flooding walkdown results may be available in early 2013, the staff will need to evaluate this information within the context of the associated plant-specific external hazards. The timeline for full implementation of the orders related to beyond-design-basis mitigation equipment (Recommendation 4.2), hardened vents (Recommendation 5.1), and spent fuel instrumentation (Recommendation 7.1), will be completed no later than December 31, 2016. Although activities associated with Recommendations 2.1 and 2.3 will be most relevant to resolution of Recommendation 3, design- and licensing-basis changes associated with beyond-design-basis external hazard mitigation equipment, boiling-water reactor venting capability, and spent fuel pool (SFP) instrumentation have the potential to impact plant-specific responses to seismically induced fires and flooding. Therefore, the staff believes that information needed to support a thorough evaluation of the Tier 3 issue for seismically induced fires and floods will not be available until the 2015 – 2016 time frame.

A complete evaluation of seismically induced fires and floods requires technical staff with significant expertise in the areas of seismic hazard evaluation, fire protection, flooding evaluation, and accident analysis. Additionally, development of a supporting PRA method requires staff with experience in seismic, fire, and flooding PRA methods. These technical areas represent critical skill sets within the staff and the availability of resources within this area is limited. Currently, staff members with the requisite expertise are engaged in other higher priority work such as National Fire Protection Association (NFPA) 805, “Performance-Based Standard for Fire Protection for Light Water Reator Electric Generating Plants,” licensing review and implementation support, external hazard standardized plant analysis risk (SPAR) model development, and the resolution of other Tier 1 and Tier 2 NTTF recommendations. Similarly, available contractors capable of supporting this work are also engaged in higher priority work activities. Therefore, staffing and contractor availability to address NTTF Recommendation 3 will be very limited over the next four years.

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[2] Recommendation 2.1 considers seismic and flood hazards separately and does not address seismically induced flooding.

Table 1. Timeline for Tier 1 NTTF Recommendations Related to Recommendation 3

Recommendation	Activity	Priority[1]	Timeline	Comments
2.1, Seismic Hazard and Risk Evaluation for Central and Eastern US (CEUS) Plants	CEUS Plants Complete Seismic Hazard Evaluation	n/a	September 2013	If higher seismic hazards relative to the design basis are identified, licensees are also to provide interim evaluation and actions taken or planned to address the higher hazard
	CEUS Plants Complete Risk Evaluation	High	October 2016	Seismic PRA is to be performed, if needed
		Low	October 2017	Seismic PRA or seismic margins analysis is to be performed, if needed.
2.1, Seismic Hazard and Risk Evaluation for Western US (WUS) Plants	WUS Plant Complete Seismic Hazard Evaluation	n/a	March 2015	If higher seismic hazards relative to the design basis are identified, licensees are also to provide interim evaluation and actions taken or planned to address the higher hazard
	WUS Plants Complete Risk Evaluation	High	April 2018	Seismic PRA is to be performed, if needed.
		Low	April 2019	Seismic PRA or seismic margins analysis is to be performed, if needed.
2.1, Flooding Hazard Evaluation	Complete Flooding Hazard Reevaluation Report	Variable	March 2013 – March 2015	If higher flooding hazards relative to the design basis are identified, licensees are also to provide interim evaluation and actions taken or planned to address the higher flooding hazard.
	Flooding Integrated Assessment Report	Variable	March 2015 – March 2017	Required only if flooding hazards higher than the design basis are identified.
2.3, Seismic and Flooding Walkdowns	Complete Walkdowns	n/a	November 2012	Walkdown procedures were issued in May 2012 (does not include inaccessible areas)
4.2, Mitigation Strategies for Beyond-Design-Basis External Events	Integrated Plan	n/a	February 2013	Addresses requirements for maintaining or restoring core cooling, containment and SFP cooling using installed equipment or resources, using portable onsite equipment and consumables, and offsite equipment. Requires providing sufficient, portable, onsite equipment and consumables, and offsite resources.
	Full Implementation		No later than Dec. 31, 2016	
5.1, Hardened Containment Vents	Integrated Plan	n/a	February 2013	Requires reliable hardened vent to remove decay heat and maintain control of containment pressure for BWR Mark 1 and Mark 2 containments.
	Full Implementation		No later than Dec. 31, 2016	
7.1, Reliable Spent Fuel Instrumentation	Integrated Plan	n/a	February 2013	Requires reliable indication of the water level in spent fuel storage pools.
	Full Implementation		No later than Dec. 31, 2016	

(1) The strategy for screening and prioritizing the site-specific seismic hazard (Recommendation 2.1) is currently under development. Once finalized, the prioritization strategy will be used to determine if a seismic margins analysis or a seismic PRA is needed to support the seismic risk evaluation.

Staff Plan

Because of staffing limitations and because significant plant-specific information related to seismic and flooding hazard evaluations will not be received for several years, the staff plans to defer the evaluation of NTTF Recommendation 3 until 2016. However, the staff also believes that some supporting information to facilitate this evaluation can be developed during the next several years. The staff plans to engage in the following activities during FY 2012 through FY 2016 to address NTTF Recommendation 3:

- Initiate the development of a PRA methodology for addressing seismically induced fires and floods. As initially described in SECY-12-0025, the staff has completed a detailed plan for developing this method (Agencywide Documents Access and Management System Accession No. ML121450222). It should be noted that the staff has limited resources available to support new fire and external hazard PRA activities. Therefore, until it obtains sufficient information from ongoing activities supporting NTTF Recommendation 2.1, 2.3, and 4.2, the staff plans to dedicate resources to this method development activity at a level that will not preclude accomplishment of other high-priority work in the fire and external hazard area (e.g., development of new NFPA 805 fire and external hazard SPAR models). The staff plans to focus method development activites in two areas:

  (1) Coordination with standards development organizations (e.g., American Society of Mechanical Engineers and American Nuclear Society) and developing more generalized approaches for assessing concurrent hazards. This will help identify the technical elements and associated high-level and supporting requirements for a suitable PRA method, and will suggest specific areas where detailed guidance is needed.

  (2) Performance of a feasibility scoping study to identify issues associated with the risk assessment of multiple concurrent hazards and evaluation of available PRA methods within this context. This study would provide information regarding the capabilities of traditional and advanced risk assessment methods (e.g., linked event tree and fault tree, dynamic simulation-based approaches) for accident scenarios where issues such as event timing, dependencies, and concurrency can influence risk significance. This study would also include an evaluation of the current state of the art for addressing seismically induced fires and floods and, more generally, concurrent hazards.

- Once it has obtained sufficient information from the Tier 1 activities identified in Table 1, the staff will re-evaluate NTTF Recommendation 3. This evaluation will be based on experience gained in developing a PRA methodology for seismically induced fires and floods and insights derived from Recommendation 2.1, 2.3, and 4.2 activities. The staff expects that this re-evaluation will result in one of the following outcomes (along with the supporting technical and regulatory basis):

  - Recommendation for regulatory action (e.g., rulemaking, order);
  - Recommendation for no regulatory action; or
  - Recommendation for further research to support future regulatory decision-making.

- The staff noted that the initial basis for identifying this Tier 3 Recommendation was associated with the limited ability of the NTTF to fully assess the basis for closure of GSI-172, because of variability or omission of critical information in the IPEEE submittals and the NTTF schedule to complete its activities. The staff considered performing a near-term activity to re-evaluate the basis for closure of GSI-172, but concluded that this historical review would add limited value toward the future safety of the nuclear power plant fleet. Additionally, given the potential for design and licensing basis changes to address external hazards over the next several years, it is likely that this ical information would no longer represent the as-built, as-operated nuclear plant fleet. Therefore, the staff does not plan to review the closure of GSI-172 so that these resources can used for higher priority work.

Schedule and Milestones

- Continue development of PRA methodology for seismically induced fires and floods as described. This will include two main subtasks:

  (1) engagement with PRA standards development organizations to develop the technical elements and requirements for the PRA method (ongoing), and

  (2) completion of a feasibility scoping study to evaluate PRA approaches for assessing multiple concurrent events (December 2014)

- Reevaluate Recommendation 3 based on information obtained from Tier 1 activities and PRA method development activities, and recommend further activities (December 2016).

Resources

Office	FY 2013		FY 2014		estimated FY 2015 and beyond	
	FTE	Dollars, $K	FTE	Dollars, $K	FTE	Dollars, $K
RES	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
NRR	[ ]	Official Use Only – Sensitive Internal Information				[ ]
NRO	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
TOTAL	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

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Tier 3 – Near-Term Task Force Recommendation 5.2

Program Plan for Reliable Hardened Vents for Other Containment Designs

Purpose

The purpose of this program plan is to provide a rationale for deferring, for the time being, further regulatory action to enhance containment overpressure protection for other containment designs (i.e., containment designs other than BWR Mark I and Mark II containments).

NTTF Recommendation and Other Direction

In SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated July 12, 2011, the NTTF recommended that the Commission direct the staff as follows:

> Reevaluate the need for hardened vents for other containment designs [i.e., containment designs other than BWR Mark I and Mark II containments], considering the insights from the Fukushima accident. Depending on the outcome of the reevaluation, appropriate regulatory action should be taken for any containment designs requiring hardened vents.

In SECY-11-0137, Recommendation 5.2 was prioritized as Tier 3 because longer term staff evaluation was required to support a decision on the need for regulatory action. In the SRM to SECY-11-0137, the Commission agreed with the Tier 3 prioritization of Recommendation 5.2. The purpose of this recommendation was to reaffirm past conclusions that hardened vents are not necessary to mitigate beyond-design-basis accident scenarios for other containment designs.

Regulations and Guidance

- 10 CFR 50.55a, "Codes and Standards," and GDC 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 require that containment be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 2 of Appendix A to 10 CFR Part 50 requires containments to be able to withstand the most severe natural phenomena such as winds, tornadoes, floods, and earthquakes and the appropriate combination of all loads.

- GDC 16, "Containment Design," of Appendix A to 10 CFR Part 50 requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

- GDC 38, "Containment Heat Removal," of Appendix A to 10 CFR Part 50 requires the provision of systems to remove heat from the reactor containment. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated

systems, the containment pressure and temperature following any loss-of-coolant accident (LOCA) and maintain them at acceptably low levels.

- GDC 50, "Containment Design Basis," of Appendix A to 10 CFR Part 50 requires containments to be designed with sufficient margin of safety to accommodate appropriate design loads.

- Paragraph (hh)(2) of 10 CFR 50.54, "Conditions of Licenses," requires licensees to develop and implement guidance and strategies to maintain or restore containment capabilities under the circumstances associated with loss of a large area of the plant as the result of explosions or fire. Expectation B.2.e of the B.5.b Phase 1 Guidance Document, dated February 25, 2002 (designated Safeguards Information), and Section 3.4.8 of the NRC-endorsed Phase 3 guidance in Nuclear Energy Institute (NEI) 06-12, Revision 2, "B.5.b Phase 2 & 3 Submittal Guidance," issued December 2006, both specify that an acceptable means of meeting the 10 CFR 50.54(hh)(2) requirements includes the development of a procedure or strategy to allow venting primary containment to secondary containment, without alternating current (ac) power, as an alternate method to remove heat from the primary containment for BWR licensees. All currently operating BWR licensees, including those with BWR Mark I, Mark II, and Mark III containment designs, adopted this approach to meeting the requirements of 10 CFR 50.54(hh)(2). There are neither current NRC regulations that require this capability for other severe (beyond-design-basis) accidents, nor design criteria for the vent paths used in this strategy.

- 10 CFR 50.34(f) requires that licensees be able to demonstrate containment integrity of applicable plants for loads associated with an accidental release of hydrogen generated from metal-water reaction of the fuel cladding, accompanied by hydrogen burning or added pressure from postaccident inerting.

- 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors," requires that licensees be able to demonstrate the structural integrity of BWRs with Mark III type containments, all pressurized-water reactors (PWRs) with ice condenser containments, and all containments used in future water-cooled reactors for loads associated with combustible gas generation.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Sections 6.2.1, "Containment Function Design," through 6.2.2, "Containment Heat Removal Systems" describe methods that are acceptable to the NRC staff for assessing the functional design of containments, their ability to withstand pressurization loads, and remove heat under post-accident conditions.

Staff Assessment and Basis for Prioritization

Over the past 30 years, the NRC has evaluated the merits of venting containments for all reactor designs. The historical focus on containment venting has been on nuclear power plants with Mark I and II containment designs. This is due, in part, to the relatively small containment

free volume of the Mark I and II containments, and the manner in which these plants respond to particular accident sequences.

A particular challenge to nuclear power plants with Mark I and II containments are postulated accident sequences in which the reactor core would be successfully cooled but the containment cooling function would be impaired (i.e., so-called “TW” sequences – or transients with loss of containment cooling). In these accident sequences, cooling of the reactor core would be achieved by transferring decay heat to the suppression pool via various safety systems. If cooling were not provided for the suppression pool in these accident sequences, the containment pressure would rise. For these reasons, a hardened vent is provided to vent the containment to prevent containment pressure from exceeding the primary containment pressure limit, while the reactor core is still being cooled. This capability was the subject of the March 12, 2012, Order EA-12-050 to nuclear power plants with Mark I and II containment designs.

Nuclear power plants with other types of containment designs are less susceptible to the same overpressurization concerns. This is because of the fundamental differences in designs and the manner in which the various designs remove decay heat from the reactor core following a transient. As an example, following a transient, PWRs transfer decay heat to steam generators, which ultimately transfer heat to the environment. As a result, reactor safety studies conducted over the years have not identified significant benefits in venting PWRs for accident sequences associated with preventing core damage.

Following a core damage accident, the containment atmosphere of all containment designs would contain radioactive material. This radioactive material could be released if a reliable hardened containment vent was opened following core damage. For this reason, the Commission directed the staff in SRM-SECY-11-0137 to evaluate the merits of new requirements that address operability of reliable hardened vents under severe accident conditions. The staff is also evaluating the merits of requiring the installation of hardened reliable containment vent filters for nuclear power plants with Mark I and II containments. Given the existing assessments of the merits of containment venting for all containment designs, the staff believes that it should continue to put a higher priority on resolving this issue for nuclear power plants with Mark I and II containment designs. Following the Commission decisions on the need for severe accident venting or filtered venting for plants with Mark I and II containments, the staff would reassess the merits of venting for other containment designs (e.g., Mark III, ice condenser, and large dry containments).

The staff’s efforts have been focused on resolving whether reliable hardened containment vents for BWR Mark I and Mark II containments should include filters and should also be operable under severe accident conditions. Given the limited resources in these specialized disciplines, the staff believes priority consideration should continue to be placed on resolving issues for BWR Mark I and Mark II containment designs. Once these issues are resolved, the staff would evaluate the merits of requiring venting for other containment designs.

Dependencies

While not dependent upon the resolution of Tier 1 Recommendation 5.1 for severe accident capable reliable hardened vents or filtered vents for Mark I and Mark II containments, the staff believes there is significant benefit to fully addressing the containment venting issue for those

plants prior to assessment of the need for reliable hardened vents for other containment designs. The staff notes that there are some dependencies between Recommendation 5.2 and Recommendation 6 (hydrogen control and mitigation). These dependencies are inherent in the manner in which the various containment designs manage the accumulation of hydrogen gas following a severe accident, and any potential impacts on the containment functions.

<u>Staff Plan</u>

The staff plans to defer consideration of venting for other containment designs (e.g., Mark III, ice condenser, and large dry containments) until the Commission reaches a decision on the need for severe accident venting and filtered venting for BWR Mark I and Mark II containments.

<u>Schedule and Milestones</u>

The staff will revise this program plan and provide the appropriate schedule and milestones following the Commission's decision on the need for severe accident venting or filtered venting for plants with BWR Mark I and Mark II containments. The staff will evaluate the technical and safety merits of venting for each particular class of containment designs, and reasonably expects that different decisions could be made for each class.

<u>Resources</u>

To facilitate budgeting, the staff has developed a budget placeholder to address the potential need for an evaluation of venting for other containment designs. As noted previously, the staff will revise this program plan, including needed resources, following the Commission's decision on the need for severe accident venting or filtered venting for plants with BWR Mark I and Mark II containments. The budget placeholder should not be viewed as a staff recommendation that action will be taken regarding the need for venting for other containment designs.

Office	FY 2013		FY 2014		estimated FY 2015 and beyond	
	FTE	Dollars, $K	FTE	Dollars, $K	FTE	Dollars, $K
RES	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
NRR	[ ]	Official Use Only – Sensitive Internal Information				[ ]
NRO	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
TOTAL	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

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Tier 3 – Near-Term Task Force Recommendation 6

Program Plan for Hydrogen Control and Mitigation Inside Containment or in Other Buildings

Purpose

The purpose of this program plan is to assess the current state of knowledge regarding hydrogen generation, transport, distribution, and combustion in light of the Fukushima Dai-ichi accident, and to determine whether any new safety issues have emerged that result in the need for additional regulatory action (e.g., rulemaking, orders).

NTTF Recommendation and Other Direction

The Task Force recommended, as part of the longer term review, that the NRC:

> Identify insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-ichi accident.

In SECY-11-0137, Recommendation 6 was prioritized as Tier 3 because longer term staff evaluation was required to support a decision on the need for regulatory action. In the SRM to SECY-11-0137, the Commission agreed with the Tier 3 prioritization of Recommendation 6.

Regulations and Guidance

- GDC 41, “Containment Atmosphere Cleanup,” of Appendix A to 10 CFR Part 50 requires, in part, that systems to control hydrogen that may be released into the reactor containment be provided as necessary to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to ensure that containment integrity is maintained.

- GDC 50, “Containment Design Basis,” of Appendix A to 10 CFR Part 50 requires, in part, the reactor containment structure to be designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin is to reflect consideration of the effects of potential energy sources that have not been included in the determination of the peak conditions, such as energy in steam generators and, as required by 10 CFR 50.44, energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling.

- 10 CFR 50.44 provides requirements applicable to the different containment designs (e.g., Mark I/II/III, ice condenser, large dry). These requirements include ensuring a mixed containment atmosphere, ensuring an inerted containment atmosphere, controlling combustible gas generated from a metal-water reaction involving a percentage of the fuel cladding surrounding the active fuel region, and monitoring oxygen and hydrogen concentrations in containment.

- 10 CFR 52.47(a)(23) and 10 CFR 52.79(a)(38) both require that design certification and combined license applicants provide in their applications a description and analysis of design features for the prevention and mitigation of severe accidents including challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass.

- RG 1.7, “Control of Combustible Gas Concentrations in Containment,” issued March 1971 and updated September 1976, November 1978, and March 2007, describes methods for implementing the requirements in 10 CFR 50.44.

- NUREG-0800, Section 6.2.5, “Combustible Gas Control in Containment,” Revision 3, issued March 2007, describes methods that are acceptable to the NRC staff for implementing 10 CFR 50.44, and Section 19.0, Revision 2, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” issued June 2007, describes acceptance criteria for determining whether an applicant has adequately demonstrated that its design properly balances preventive and mitigative (e.g., hydrogen generation and control) features.

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

The NRC has long recognized the potential impact on nuclear safety and risk from the generation and combustion of hydrogen gas from severe accidents. In addition to the accident at Fukushima Dai-ichi, hydrogen was also generated and burned during the Three Mile Island (TMI) accident in 1979. Following the TMI accident, the NRC strengthened the hydrogen control safety requirements, and initiated numerous broad multi-year research programs to evaluate core melt progression and a range of severe accident issues, including hydrogen generation and control. As a result, the NRC has a significant knowledge base regarding hydrogen generation, transport, distribution, and combustion, and has incorporated that knowledge base into the agency’s analytical tools, and deterministic and probabilistic models to assess the safety and risk implications of hydrogen generation and combustion. Also, over the years, the NRC has identified many Generic Safety Issues and Generic Issues (e.g., A-48, B-14, 106, 121, 167, 189, 195, 198) involving hydrogen and has evaluated their impact on nuclear safety. Given the significance of the events at Fukushima Dai-ichi, the staff concludes that it is appropriate to assess the existing state of knowledge regarding hydrogen generation and combustion in light of the Fukushima Dai-ichi accident, and the potential impact on reactor safety and risk to determine whether any additional regulatory action is needed.

In 2003, the NRC revised the hydrogen control requirements in 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Reactors.” This rule eliminated the requirements for hydrogen recombiners and hydrogen purge systems in currently licensed light water reactors, and relaxed the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their risk significance. However, the rule retained existing requirements for ensuring a mixed atmosphere, inerting Mark I and II containments, and providing a hydrogen control system capable of accommodating an amount of hydrogen generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region in Mark III and ice condenser containments. The technical bases for the regulations were established from

experience at TMI along with bounding estimates for the amount of hydrogen likely to be generated by a severe core damage accident.

This rule also specifies requirements for combustible gas control in future water-cooled reactors which are similar to the requirements specified for existing plants. However, a key difference is the need to accommodate an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction. Particularly, if a containment does not have an inerted atmosphere, it must limit hydrogen concentrations in containment during and following an accident that releases hydrogen (equivalent to 100 percent fuel-coolant reaction) when uniformly distributed to less than 10 percent (by volume); and maintain containment structural integrity and appropriate accident mitigating features.

Boiling water reactors with Mark I or Mark II type containments must have an inerted atmosphere. This concept reduces oxygen sufficiently to suppress combustion; thereby a hydrogen generation limit is not specified. The result of a hydrogen combustion event is characterized as a relatively sharp pressure pulse and thus the intent of the hydrogen rule precludes this occurrence inside containment; but, it does not recognize the slow build-up of containment pressure as a result of the hydrogen gas generated by postulated severe core damage accidents. Therefore, containment pressure control is addressed in the severe accident management guidelines. Essentially, pressure control for severe accidents in Mark I and II containments is also related to the control of hydrogen and other non-condensable gases for the containment and now, based on the experience from the Fukushima Dai-ichi accident, can have an impact on the reactor building.

The key attributes of Mark I and Mark II containments designs are: (1) the containment free gas volumes are relatively small compared to other light water reactors, so the hydrogen gas (and other non-condensable gases produced) and steam buildup in containment will affect the pressure rise more dramatically, and (2) BWR reactor cores have about three times the zirconium inventory compared to comparable PWR power levels, so there is a greater potential to generate significant amounts of hydrogen gas which also will affect adversely the containment pressures. Consequently, a reliable containment venting strategy is needed and must be properly integrated in the severe accident management guidelines. Thus, the issues of containment pressure control and hydrogen gas control for Mark I and II containments are more linked than they are for other containment designs.

BWR facilities with Mark III containments and PWR facilities with ice condenser containments meet the requirements of 10 CFR 50.44 for hydrogen control systems by providing hydrogen igniters inside containment to control the buildup of hydrogen gas. These igniters are operated in two redundant trains, with each train powered by one of the redundant safety-grade ac electrical power systems. The emergency operating procedures direct energizing the igniters to ensure hydrogen concentrations remain within acceptable limits for maintenance of containment integrity.

GSI-189, “Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion during a Severe Accident,” raised questions about the effectiveness of these igniter systems during a prolonged station blackout (SBO) scenario. In response to the issues raised in GSI-189, licensees operating BWRs with Mark III containments voluntarily provided equipment, procedures, and training to support provision of nonsafety-grade backup electrical power from portable generators to one train of the igniters that is independent of much

of the safety-grade ac and dc onsite power systems. Licensees operating PWRs with ice condenser containments voluntarily developed similar capabilities, but with less independence from the on-site ac and dc power distribution systems.

Because Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," requires licensees to develop strategies that are capable of mitigating the effects of a prolonged SBO scenario, the staff has proposed in the Draft Japan Lessons-Learned Project Directorate Interim Staff Guidance (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," the staff position that licensees with installed hydrogen igniters shall develop and maintain strategies to provide alternative power from generating equipment independent of the safety-related on-site power sources to supply electricity to one train of hydrogen igniter equipment. Independent alternative power generating equipment shall be accessible and capable of installation in the transition phase. These requirements have been adopted and incorporated into Nuclear Energy Institute (NEI) 12-06, Revision C, 'Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,' dated July 2012.

To determine whether additional regulatory action is needed to resolve Recommendation 6, the staff has developed this project plan to address the following key questions:

(1) Did the accident reveal new information on the manner in which hydrogen gas is generated, transported, distributed, and combusts?

(2) Was the manner in which the containment and the reactor buildings failed in the accident consistent with our understanding?

(3) Did the accident reveal important gaps in our deterministic and probabilistic risk assessment modeling of the threat from hydrogen generation, distribution, and combustion?

(4) Did the accident reveal new information that conflicts with the technical basis for the existing hydrogen generation and control requirements?

(5) Did the accident reveal new technical information, not previously considered, to pursue regulatory action?

<u>Dependencies</u>

The generation and combustion of hydrogen gas during a severe accident is dependent upon many variables, such as the nuclear power plant design, particular severe accident sequence consideration, and the plant's emergency response to any hydrogen produced. The NTTF recognized such, and identified some of the dependencies between Recommendation 6 and the other NTTF Recommendations. In particular, the NTTF noted that:

> Implementation of Task Force Recommendation 4, associated with prolonged SBO, would reduce the likelihood of core damage and hydrogen production. In addition, implementation of Recommendation 5 to enhance the containment venting capabilities for Mark I and Mark II containments, while primarily intended for overpressure protection,

would also provide for the reliable venting of hydrogen to the atmosphere. These two steps would greatly reduce the likelihood of hydrogen explosions from a severe accident.

As a result, in addressing Recommendation 6, the staff concludes that it is important to coordinate efforts in addressing and resolving this recommendation with the other NTTF recommendations and to recognize the synergy in the integrated plant's safety response. Specifically, the direction from the Commission on the forthcoming SECY paper describing potential options for additional performance requirements for reliable, severe accident capable, hardened containment vents (including consideration of filters) will have a definitive impact on the path forward and level of effort associated with Recommendation 6.

Staff Plan

The staff plans to undertake the following tasks:

(1) Assess additional hydrogen control measures and potential hydrogen ingress into adjacent buildings – including the feasibility, safety significance, and risk implications of providing additional hydrogen control measures for the primary containment and connected structures, (e.g., reactor buildings and auxiliary buildings). The staff's approach will be to conduct stakeholder meetings and evaluate additional mitigative measures to improve the capability of reactor and auxiliary buildings to deal with hydrogen released during a severe accident. The staff will also quantify the impact on safety and risk of the mitigative measures and address uncertainty in performance.

(2) Evaluate the Fukushima Dai-ichi accident sequences with particular emphasis on hydrogen generation from all sources and timing. The staff will compare the accident timing and amount of hydrogen generated both in-vessel and ex-vessel, to that predicted in comparable severe accident scenarios for U.S. nuclear power plants. This assessment will include a review of information from sources such as the Government of Japan, Tokyo Electric Power Company (TEPCO), Institute of Nuclear Power Operations (INPO), and international organizations, along with any analytical modeling predictions such as the Department of Energy (DOE)/NRC MELCOR forensics analysis and EPRI MAAP analysis.

(3) Review information that becomes available in the near term on potential containment release pathways (e.g., upper drywell head, equipment/personnel hatches, instrument penetrations, bellows, seals) for hydrogen ingress into the reactor building. The staff will also follow the hydrogen combustion assessments on the reactor building and the safety related equipment. This assessment will include a review of information from sources such as the Government of Japan, TEPCO, INPO, and international organizations, along with any analytical modeling predictions such as the DOE/NRC MELCOR forensics analysis and EPRI MAAP analysis. This task also includes the potential need for new analytical modeling and experimental evaluation of possible containment release pathways. The staff notes uncertainties in the generation (e.g., from steel oxidation in-vessel), transport, distribution, and migration of hydrogen gas during and following a severe accident due in part to limitations in analytical modeling capabilities and also the complex layout of nuclear power plants. Also, multiple reactor coolant system gas/steam release locations are likely in a variety of severe accident sequences.

(4) Assess the technical basis for NRC's existing hydrogen generation and control requirements in 10 CFR 50.44 against the results of Tasks 1-3 above. The focus of the assessment is to confirm the validity of the existing technical basis or identify gaps, and characterize the safety and risk significance of any identified gaps.

The staff will integrate the results of the above tasks into a final report that will be used to determine whether any additional regulatory action is needed to address Recommendation 6. In carrying out the above assessments, the staff is limited by the extent of information available and the validity of that information in determining the plant's response to the Fukushima Dai-ichi accident. The July 2011 NTTF Report noted that sufficient information was not yet available for the Task Force to reasonably formulate any further specific recommendations related to combustible gas control based on insights from the Fukushima Dai-ichi accident. While a year has passed since the Task Force issued its report, in actuality, very little additional empirical information has been reported regarding hydrogen generation and control from the accident. Recent efforts and information released over the past year have focused on analytical modeling and predictions that provide some insights into the plant's response. However, the actual data and conclusive evidence on the timing and amount of hydrogen generation, including contributions from both in-vessel and potential ex-vessel reactions, and the physical mechanisms and pathways for the hydrogen release from the containment to the reactor building may not be available for many years. Experience with the TMI-2 accident indicates that a thorough accident investigation, including physical investigation and deconstruction of the reactor building, containment, and reactor pressure vessel is necessary to fully understand and learn from a severe accident.

The staff also notes that hydrogen generation and control following a severe accident and the resulting impacts on a nuclear power plant are a highly specialized technical discipline. While the NRC and international organizations had substantial hydrogen research and regulatory programs in place in the 1980s and 1990s, those resources and many of the experts working in that discipline are no longer available.

Schedule and Milestones

- Engage stakeholders as necessary to support Task 1 – 2nd Quarter FY 2013
- Complete Task 1 Final Report – 1st Quarter FY 2014
- Complete Task 2 Final Report on hydrogen generation and timing – 1st Quarter FY 2014
- Complete Task 3 Final Report on hydrogen transport and distribution – 1st Quarter FY 2014
- Assess the results of Final Reports on Tasks 1-3 and complete Task 4 Final Report – 2nd Quarter FY 2014

Resources

Office	FY 2013		FY 2014		estimated FY 2015 and beyond	
	FTE	Dollars, $K	FTE	Dollars, $K	FTE	Dollars, $K
RES	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
NRR	[ ]	Official Use Only – Sensitive Internal Information				[ ]
NRO	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
TOTAL	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

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Tier 3 – Near-Term Task Force Recommendations 9.1, 9.2, 9.3, 10.1, 10.2, 10.3, 11.1, 11.2, 11.3, and 11.4

Program Plan for Emergency Preparedness (EP) Enhancements for Prolonged SBO and Multiunit Events, Emergency Response Data System Capability, Additional EP Topics for Prolonged SBO and Multiunit Events, EP Topics for Decisionmaking, Radiation Monitoring, and Public Education

Purpose

The purpose of this program plan is to evaluate various emergency preparedness (EP) recommendations identified by the NTTF and determine whether there is a sufficient technical basis to support rulemaking.

NTTF Recommendation and Other Direction

In Recommendations 9, 10, and 11, the Task Force recommended that the NRC:

9.1 Initiate rulemaking to require EP enhancements for multiunit events in the following areas:

- personnel and staffing
- dose assessment capability
- training and exercises
- equipment and facilities

9.2 Initiate rulemaking to require EP enhancements for a prolonged Station Blackout (SBO) in the following areas:

- communications
- Emergency Response Data System (ERDS) capability
- training and exercises
- equipment and facilities

9.3 Order licensees to do the following until rulemaking is complete in the following areas:

- Maintain ERDS capability throughout the accident.

10.1 Analyze current protective equipment requirement for emergency responders and guidance based upon insights from the accident at Fukushima.

10.2 Evaluate the command and control structure and the qualifications of decisionmakers to ensure that the proper level of authority and oversight exists in the correct facility for a long-term SBO or multiunit accident or both. The evaluation should consider concepts such as whether decisionmaking authority is in the correct location (i.e., at the facility), whether currently licensed operators need to be integral to the ERO outside of the control room (i.e., in the TSC), and whether licensee emergency directors should have a formal “license” qualification for severe accident management.

10.3 Evaluate ERDS to do the following:

- 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.

- Determine whether the data set currently being received from each site is sufficient for modern assessment needs.

- Determine whether ERDS should be required to transmit continuously so that no operator action is needed during an emergency.

11.1 Study whether enhanced onsite emergency response resources are necessary to support the effective implementation of the licensees' emergency plans, including the ability to deliver the equipment to the site under conditions involving significant natural events where degradation of offsite infrastructure or competing priorities for response resources could delay or prevent the arrival of offsite aid.

11.2 Work with FEMA, States, and other external stakeholders to evaluate insights from the implementation of EP at Fukushima to identify potential enhancements to the U.S. decisionmaking framework, including the concepts of recovery and reentry.

11.3 Study the efficacy of real-time radiation monitoring onsite and within the EPZs (including consideration of AC independence and real-time availability on the internet).

11.4 Conduct training, in coordination with the appropriate Federal partners, on radiation, radiation safety, and the appropriate use of KI in the local community around each nuclear power plant.

In SECY-11-0137, these Recommendations (i.e., 9.1, 9.2, 9.3 addressing ERDS capability, 10.1, 10.2, 10.3, 11.1, 11.2, 11.3, and 11.4) were prioritized as Tier 3 because they were dependent upon the availability of critical skill sets or required longer term staff evaluation to support a decision on the need for regulatory action. In the SRM to SECY-11-0137, the Commission agreed with the Tier 3 prioritization of these EP Recommendations.

In addition to the Tier 3 EP recommendations discussed above, the staff is continuing to evaluate the following Tier 2 EP recommendations:

9.3 Order licensees to do the following until rulemaking is complete:

- Add guidance to the emergency plan that documents how to perform multiunit dose assessment (including releases from the spent fuel pool(s)) using the licensee's site-specific dose assessment software and approach.

- Conduct periodic training and exercises for multiunit and prolonged SBO scenarios. Practice (simulate) the identification and acquisition of offsite resources, to the extent possible.

- Ensure EP equipment and facilities are sufficient for dealing with multiunit and prolonged SBO scenarios.

To address these Tier 2 and 3 EP issues, the staff developed this integrated program plan. This plan also provides an update on the Tier 1 action associated with NTTF Recommendation 9.4.

9.4 Order licensees to complete the ERDS modernization initiative by June 2012 to ensure multiunit site monitoring capability.

Regulations and Guidance

- 10 CFR 50.47, “Emergency Plans,” includes 16 EP planning standards.
  10 CFR Part 50, Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” describes information needed to demonstrate compliance with the EP requirements.

- NUREG-0654, Revision 1, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” issued November 1980, provides guidance and an acceptable means for demonstrating compliance with the EP regulations.

- NUREG-0696, “Functional Criteria for Emergency Response Facilities,” issued February 1981, provides guidance that can be used to describe the facilities and systems that licensees should use for emergency response to accidents, such as the technical support system, operational support center, and emergency offsite facility.

Staff Assessment and Basis for Prioritization

The staff considers the existing EP framework of regulations and guidance provides reasonable assurance of adequate protection of public health and safety in a radiological emergency. The staff has conducted several studies that have informed the NRC evaluation of the adequacy of this approach. The results of these studies have been published in the following documents: (1) NUREG/CR-6953, “Review of NUREG-0654, Supplement 3, ‘Criteria for Protective Action Recommendations for Severe Accidents’,” evaluated the efficacy of various protective action strategies within the EPZ (ADAMS Accession Nos. ML080360602, ML083110406, and ML102380087); (2) NUREG/CR-6864, “Identification and Analysis of Factors Affecting Emergency Evacuations,” issued January 2005, examined large evacuations in the United States between 1990 and 2003 to more fully understand the dynamics involved (ADAMS Accession Nos. ML050250245 and ML050250219); and (3) Draft NUREG-1935, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Report,” issued January 2012 (ADAMS Accession No. ML120250406), evaluated hypothetical evacuations within EPZs and beyond in response to a series of accident scenarios. These studies have informed the NRC’s conclusion that the NRC’s existing EP framework provides reasonable assurance of adequate protection of public health and safety in the event of a radiological emergency at a U.S. power reactor facility.

As noted in SECY-11-0137, the staff proposed that the existing EP regulations and guidance be reviewed for enhancements to address multiunit and SBO events; enhancements to ERDS

capability; and enhancements to EP decisionmaking, radiation monitoring, and public education practices. In the NTTF Report, the Task Force concluded that continued operation and continued licensing activities do not pose an imminent risk to the public health and safety and are not inimical to the common defense and security. Therefore, the staff has concluded that, given the above, as well as the enhancements required by the recent revisions to the EP regulations and the proposed enhancements to licensee staffing and communication, these resulting Tier 2 and 3 issues do not require immediate regulatory action. As a result, the staff plans to include the following Tier 2 and 3 issues in a single ANPR.

NTTF Recommendations 9.1 and 9.2 are similar. The recommendations of the NTTF state that rulemaking is necessary to expand elements of emergency plans to ensure that licensees can respond effectively to SBO and multiunit events. The use of an ANPR will allow the staff to engage stakeholders and solicit input to inform the development of a technical basis to address EP enhancements pertaining to training, drills and exercises, equipment and facilities, and the ERDS. In addition, the staff has requested that licensees respond to two 10 CFR 50.54(f) letters regarding staffing and communications and will use the licensee's responses to the 10 CFR 50.54(f) letters to inform the development of the ANPR.

NTTF Recommendation 9.3 generally aligns in terms of issues with 7 of the 16 existing EP planning standards including staffing (10 CFR 50.47(b)(2)), equipment/facilities (10 CFR 50.47(b)(8)), assessment (10 CFR 50.47(b)(9)), training (10 CFR 50.47(b)(15)), exercises (10 CFR 50.47(b)(14)), offsite resources (10 CFR 50.47(b)(3)), and communications (10 CFR 50.47(b)(6)). However, the planning standards were developed for a single unit. The staff will focus its efforts on how licensees address these seven planning standards given an SBO and multiunit accidents.

Licensees have implemented Recommendation 9.4. An ERDS modernization initiative was completed in June 2012, and all licensees are transmitting ERDS information via a virtual private network. The NRC now has the ability to receive and process ERDS data from all power reactor units. The staff will request stakeholder feedback on the need for a regulatory requirement to supply emergency power to the equipment and systems used to collect and transmit the ERDS data to the NRC.

NTTF Recommendations 9.2, 9.3, and 10.3 focus on several aspects of the ERDS (i.e., alternate methods to transmit ERDS data, data point sufficiency, and continuous ERDS transmission) and are similar. The staff determined that these aspects of the ERDS may need a more integrated and comprehensive set of requirements. Specifically, the areas of focus will be (1) to determine whether licensees need to evaluate existing reactor safety parameters and provide the NRC additional data points, and (2) to determine whether licensees need to change the methods of transmitting ERDS data to address the issues identified in the NTTF report and in previous lessons-learned documents. The staff will include these items in the ANPR and will request stakeholder feedback on the need for regulatory requirements.

NTTF Recommendations 10.1, 10.2, and 11 are similar, and the staff has determined that these aspects of EP may also need a more integrated and comprehensive set of requirements. Specifically, the NTTF recommended (1) reanalyzing current protective equipment requirements for emergency responders, (2) evaluating the command and control structure and qualifications of decisionmakers to ensure that the proper level of authority and oversight exists at the correct facility for a long-term SBO and/or multiunit event, (3) conducting additional studies to determine

whether enhanced onsite emergency response resources are necessary to support effective implementation of the licensees' emergency plans including delivery of equipment on site under degraded conditions and competing priorities, (4) working with FEMA to identify potential enhancements to the U.S. decisionmaking framework, including recovery and reentry to a site, (5) studying the efficacy of real-time radiation monitoring on site and within the EPZs (including consideration of ac independence and real-time availability on the Internet), and (6) conducting training, in coordination with Federal partners, on radiation safety and the appropriate use of KI. The staff will include these items in the ANPR and will request stakeholder feedback on the need for regulatory requirements.

Dependencies

The staff expects to use information gained from the responses to the Tier 1 EP issues that resulted in issuance of a 50.54(f) letter regarding staffing and communications.

Staff Plan

The staff plans to use a single ANPR to collect more information regarding NTTF Recommendations 9, 10, and 11. The staff's planned approach includes the following:

(1) Issue an ANPR to engage stakeholders on rulemaking activities associated with the methodology for integration of onsite emergency response processes, procedures, the ERDS, and training and exercises. Interact with stakeholders to inform the modifications to EP guidelines and/or equipment capabilities that would include guidance for multiunit and SBO events in an integrated manner, and to clarify command and control issues, as appropriate.

(2) Evaluate licensee responses to the 10 CFR 50.54 letter on staffing and communications for EP, and take regulatory action to require implementation, as appropriate.

Schedule and Milestones

(1) Develop and issue final rule - 4.25 years. The staff plans to initiate work on the ANPR in FY 2014; however, additional resources would be needed to complete the activites as outlined in the schedule.

   a. Develop and issue an ANPR to obtain stakeholder input—4 months.

   b. Develop a regulatory basis incorporating stakeholder feedback—nominally, 13 months following initiation of the action.

   c. Issue a proposed rule and supporting guidance for comment—nominally, an additional 16 months following completion and acceptance of the regulatory basis (incorporates 4 months for Commission review and staff response to the SRM).

   d. Meet with the Advisory Committee on Reactor Safeguards (ACRS) during the proposed rule stage (if requested by ACRS).

e. Issue final rule and supporting guidance—nominally, an additional 22 months (including a 75-day public comment period, 4 months of Commission review, staff response to the final rule SRM, 3 months for Office of Management and Budget approval for final rule, and a meeting with ACRS).

(2) Licensing activities – Schedule to be determined, dependent on rule requirements.

a. licensee submittals

b. staff reviews

(3) Inspection activities – Schedule to be determined, dependent on licensee implementation timeframe.

a. Incorporate inspection into the Reactor Oversight Process

b. Conduct inspections and document results

<u>Resources</u>

Office	FY 2013		FY 2014		estimated FY 2015 and beyond	
	FTE	Dollars, $K	FTE	Dollars, $K	FTE	Dollars, $K
NRR	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
NRO	[ ]	Official Use Only – Sensitive Internal Information				[ ]
NSIR	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
OGC	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
TOTAL	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

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Tier 3 – Near-Term Task Force Recommendation 12.1

Program Plan for Reactor Oversight Process Modifications To Reflect the Recommended Defense-in-Depth Framework

Purpose

The purpose of this program plan is to discuss how the staff will consider the resolution of Recommendation 1 (i.e., establish a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations) to modify the Reactor Oversight Process (ROP).

NTTF Recommendation and Other Direction

The Task Force recommended that the NRC expand the scope of the annual ROP self assessment and biennial ROP realignment to more fully include defense-in-depth considerations.

In SECY-11-0137, Recommendation 12.1 was prioritized as Tier 3 because it was dependent upon Recommendation 1. In the SRM to SECY-11-0137, the Commission agreed with the Tier 3 prioritization of Recommendation 12.1.

Regulations and Guidance

- Management Directive (MD) 8.13, "Reactor Oversight Process" contains the ROP policy, including discussion of regulatory framework and the seven cornerstones of safety.

- MD 8.14, "Agency Action Review Meeting" provides a description of the annual meeting of senior NRC management to review ROP actions.

- Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program" describes the NRC's assessment process for inspection findings and performance indicators, including agency response to these items (e.g., the Action Matrix).

- IMC 0350, "Oversight of Operating Reactors in a Shutdown Condition with Performance Problems" establishes criteria for the oversight of licensee performance for licensees that are in a shutdown condition as a result of significant performance problems or operational event.

- IMC 0307, "Reactor Oversight Process Self-Assessment Program" discusses the ROP self-assessment program and the evaluation of the overall effectiveness of the ROP through its success in meeting its preestablished goals and intended outcomes.

- IMC 0308, "Reactor Oversight Process (ROP) Basis Document" describes the basis for the significant decisions reached by the NRC staff during the development and implementation of the ROP.

- SECY-01-0111, "Development of an Industry Trends Program for Operating Power Reactors"
- SECY-00-0061, "Proposed Revision to the Enforcement Policy to Address the Revised Reactor Oversight Process"
- SECY-00-0049, "Results of the Revised Reactor Oversight Process Pilot Program," dated February 24, 2000
  - SRM Part 1 for SECY-00-0049, dated March 8, 2000
  - SRM Part 2 for SECY-00-0049, dated May 17, 2000

Staff Assessment and Basis for Prioritization

The staff will continue to implement the ROP in accordance with current policy until such time that an action plan for Recommendation 1 has been established. At that time, the staff will begin to consider potential changes to the ROP self assessment and realignment programs.

Dependencies

As discussed in SECY-11-0137, Recommendation 12.1 is dependent on Recommendation 1 (i.e., establish a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations).

Staff Plan

The staff will defer action on Recommendation 12.1 until the Commission has provided staff guidance regarding Recommendation 1.

Schedule and Milestones

The schedule for the resolution of Recommendation 1 is for the staff to provide the Commission with a notation vote paper in February 2013. Once the Commission provides staff guidance on Recommendation 1, the staff will initiate a review of potential enhancements to the ROP.

It is anticipated that the staff would be ready to update the Commission on the status of Recommendation 12.1 in the 2013 annual ROP Self-assessment SECY paper (issued in spring 2014).

Periodic stakeholder interactions will take place as necessary during the NRC's routine monthly meetings with NEI and the industry on ROP topics.

Resources

The staff does not intend to expend any resources on Recommendation 12.1 until an action plan has been established for Recommendation 1.

Resources will remain the same for any potential change in annual inspection activities. This would be accomplished by including any new inspection samples into the existing baseline

inspection program. Any potential one time confirmatory inspections would be accomplished through the already funded generic inspection process (Temporary Instructions). Any inspections required to assess the implementation of Tier 1 recommendations would be handled in a similar manner.

Any potential modifications to the ROP self assessment and inspection programs and procedures would be made using the current funding for the ROP and inspection program activities (i.e., the realignment processes).

Tier 3 – Near-Term Task Force Recommendation 12.2

Program Plan for Staff Training on Severe Accidents and Resident Inspector Training on Severe Accident Mitigation Guidelines

Purpose

The purpose of this program plan is to describe the current level of NRC staff training conducted related to severe accidents and severe accident management guidelines, and to outline the considerations for future training enhancements.

NTTF Recommendation and Other Direction

The Task Force recommended that the NRC enhance NRC staff training on severe accidents, including training resident inspectors on severe accident management guidelines (SAMGs).

In SECY-11-0137, Recommendation 12.2 was prioritized as Tier 3 because it is dependent on the resolution of Recommendation 8.

Regulations and Guidance

The following probabilistic risk assessment (PRA) and reactor technology courses are currently available for NRC staff training on severe accident progression, consequences, and emergency operating procedures (EOPs):

- Accident Progression Analysis (P-300)
- Accident Consequence Analysis (P-301)
- Perspectives on Reactor Safety (R-800)
- General Electric Technology Emergency Procedure and Severe Accident Guidelines (R-624B)
- Westinghouse Emergency Procedure Guidelines (R-624P)
- Babcock and Wilcox Emergency Procedure Guidelines (R-326C)
- Combustion Engineering Emergency Procedure Guidelines (R-325C)

The Office of Nuclear Regulatory Research sponsored a 2-day training seminar on SAMGs which was offered by the Pressurized Water Reactors Owners Group. A video of the training is available internally to the staff at http://r2.nrc.gov/videoarchive/ViewVideo.cfm?vlink=267.

Staff Assessment and Basis for Prioritization

The nuclear industry developed SAMGs during the 1980s and 1990s in response to the Three Mile Island accident. This effort included extensive research and analysis, including PRAs, on

severe accidents and severe accident phenomena. SAMGs are meant to enhance the ability of the operators to manage accident sequences that progress beyond the point where EOPs and other plant procedures are applicable and useful. In NRC Generic Letter 88-20, Supplement 2, "Accident Management Strategies for Consideration in the Individual Plant Examination Process," dated April 4, 1990, the NRC encouraged, but did not require, licensees to develop and implement SAMGs. Consequently, only limited NRC staff and inspector training courses on severe accidents and SAMGs are currently available.

The scope of enhancements to NRC staff training on severe accidents, including SAMGs is dependent on the resolution of Recommendation 8. However, because an understanding of severe accident progression and consequences is fundamental to the understanding of SAMGs, the NRC staff training on severe accident progression and consequences can be conducted in parallel with activities to resolve Recommendation 8. In addition, enhancements to existing training courses to include lessons-learned from the Fukushima accident could be initiated in the near-term. Longer-term enhancements to training could be informed by the SOARCA study and follow-up research activities, the Level 3 PRA study, and future Fukushima lessons-learned.

Currently, NRC staff training on severe accidents is required by a limited number of qualification programs (e.g., senior reactor analyst (IMC-1245, App. C-9), reactor technical reviewers and risk analysts (NRR ADM-504, branch-specific training plans), nuclear safety professional development program participants (NRR ADM-502, reactor track), and construction reactor oversight specialist (NRO-PER-105)).

<u>Dependencies</u>

As noted above, full consideration of Recommendation 12.2 is dependent upon resolution of Recommendation 8. Nevertheless, there are activities that the staff will undertake that are supportive of implementation of Recommendation 12.2.

<u>Staff Plan</u>

The staff plans to seek opportunities in the near term to enhance staff training in severe accident progression and consequences. This includes the following:

- Review the frequency of existing severe accident training courses for consideration of offering the courses on a greater frequency, including the export of severe accident courses to regional offices.

- Enhance the existing severe accident courses such as, R-800, to include lessons-learned from the Fukushima accident.

- Evaluate adequacy of existing NRC qualification programs for required training on severe accidents.

- Interact with stakeholders to inform the enhancements to training on severe accidents.

As a longer-term action, the staff plans to re-evaluate additional enhancements to training in parallel with the resolution of Recommendation 8. The staff does not plan consolidation of this recommendation with any other NTTF recommendations.

Schedule and Milestones

(1) Implement near-term enhancements to existing NRC severe accident training. This will be conducted in parallel with resolution of Recommendation 8 and is dependent on updating the R-800 course.

   a. Increase frequency of severe accident training, including exporting training to the Regions.

   b. Update existing severe accident training to include Fukushima lessons-learned.

   c. Revise NRC qualification program training requirements for severe accidents.

   d. Meet with stakeholders to inform training enhancements.

(2) Implement longer-term enhancements to NRC staff training on severe accidents, including resident inspector training on SAMGs, dependent on resolution of Recommendation 8 (approximately 2016).

   a. Enhance training based on SOARCA study and follow-up research activities, the Level 3 PRA study, and longer-term Fukushima lessons-learned.

   b. Revise NRC qualification training to include training on SAMGs.

   c. Develop new training courses on severe accident progression and consequences and SAMGs, as determined to be necessary.

   d. Meet with stakeholders to inform training enhancements.

<u>Resources</u>

Office	FY 2013		FY 2014		estimated FY 2015 and beyond	
	FTE	Dollars, $K	FTE	Dollars, $K	FTE	Dollars, $K
NRR	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
NRO	[ ]					[ ]
NSIR	[ ]					[ ]
OCHCO	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
TOTAL	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

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Notes:

1. The resources necessary for development of longer-term enhancements to training is dependent on the resolution of Recommendation 8.
2. The resources for training of NRC staff are included in the inspection resource estimates provided in SECY-11-0137 for Recommendation 8 ([ ] total estimated FTE). OUO
3. The staff anticipates that enhancement of training will involve NRC staff in the areas of Human Factors, Operator Licensing, Incident Response/Emergency Preparedness, Inspection, Nuclear Engineering, Thermal Hydraulics, and Materials Engineering.

Tier 3 – SECY-11-0137 Additional Recommendation 3

Program Plan for Basis of Emergency Planning Zone Size

Purpose

The purpose of this program plan is to define the activities that the staff will undertake to evaluate the basis of the emergency planning zone size.

NTTF Recommendation and Other Direction

In SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, the staff identified a number of additional issues with a clear nexus to the Fukushima Dai-ichi event that may warrant regulatory action, but which were not included in the NTTF recommendations. One of the additional recommendations was to evaluate the basis of the plume exposure pathway Emergency Planning Zone (EPZ) size.

Regulations and Guidance

- 10 CFR 50.47(c)(2) "Emergency plans," states that the plume exposure pathway EPZ for nuclear power plants shall, generally, consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.

- NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, issued November 1980, describes guidance and an acceptable means for demonstrating compliance with the Commission's regulations.

- 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.

Staff Assessment and Basis for Prioritization

Following the event at Fukushima, the NRC in conjunction with other U.S. Government entities issued a prudent, conservative travel advisory for American citizens within a 50-mile range of the Fukushima plant. 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 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 (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 plants. Any changes to EPZs will be reviewed in coordination with FEMA.

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'," evaluates the efficacy of various protective action strategies within the EPZ (ADAMS Accession Nos. ML080360602, ML083110406, and ML102380087). NUREG/CR-6864, "Identification and Analysis of Factors Affecting Emergency Evacuations" examines large evacuations in the United States between 1990 and 2003 to gain a fuller understanding of the dynamics involved (ADAMS Accession Nos. ML050250245 and ML050250219). Draft NUREG-1935, "State of the Art Reactor Consequence Analysis" evaluates hypothetical evacuations within EPZs and beyond in response to a series of accident scenarios (ADAMS Accession No. ML120250406). 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, in a 1979 policy statement (44 FR 61123, October 23, 1979), NRC endorsed NUREG-0396, which provides bases for the 10-mile EPZ, including an assumption that the planning conducted for 10 miles would provide a substantial basis for expansion of the EPZ should it ever be necessary.

In the coming years, there are extensive plans to further study the potential health effects for the released radioactivity from the Fukushima site. The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) plans a two-year assessment of Fukushima impacts; and a major initiative is planned, the Fukushima Health Survey, that will inform future more detailed dose assessments by recreating the whereabouts of every Fukushima prefecture resident from the time of the March 11 nuclear accident onwards. The NRC staff will continue to monitor the results of these efforts, and their potential implications regarding the U.S. regulatory approach to emergency planning around nuclear power plants, including the EPZ size. In addition, the NRC is conducting a Level 3 Probabilistic Risk Assessment (PRA) to gain a better

understanding of potential radiological effects of postulated accident sequences including sites with multiple units.

Dependencies

The staff has not identified any dependencies on other NTTF Recommendations.

Staff Plan

The staff believes that the existing basis for the EPZ size remains valid (including for multiunit events). With regard to this recommendation, the staff plans a longer-term action that is already being evaluated by existing activities. The staff will use insights from the current Level 3 PRA study as well as information obtained from the UNSCEAR assessment to inform the evaluation of the potential impacts that a multiunit event may have on the EPZ. Such insights would be primarily related to understanding the sensitivity of the calculated offsite health consequences to various modeling assumptions.

Schedule and Milestones

The staff will use the results of the Level 3 PRA and the UNSCEAR assessment to inform the process of evaluating the EPZ basis. This effort could take 3 – 4 years to complete.

Resources

Office	FY 2013		FY 2014		estimated FY 2015 and beyond	
	FTE	Dollars, $K	FTE	Dollars, $K	FTE	Dollars, $K
NSIR	[ ]					[ ]
TOTAL	[ ]					[ ]

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Tier 3 – SECY-11-0137 Additional Recommendation 4

Program Plan for Prestaging of Potassium Iodide Beyond 10 Miles

Purpose

The purpose of this program plan is to define the activities that the staff will undertake to evaluate whether potassium iodide should be pre-staged beyond the current 10-mile zone.

NTTF Recommendation and Other Direction

In SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, the staff identified a number of additional issues with a clear nexus to the Fukushima Dai-ichi event that may warrant regulatory action, but which were not included in the NTTF recommendations. One of the additional recommendations that warranted further consideration and potential prioritization was to evaluate whether Potassium Iodide (KI) should be pre-staged beyond the current 10 mile zone.

Regulations and Guidance

The regulation in 10 CFR 50.47(b)(10) "Emergency plans," states that the onsite and offsite emergency response plans for nuclear power reactors must meet specified standards, one of which is a range of protective actions developed for the plume exposure pathway EPZ for emergency workers and the public. In developing this range of actions, consideration is given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of KI, as appropriate. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale are developed.

Staff Assessment and Basis for Prioritization

Following the event at Fukushima, stakeholders have submitted comments to the NRC requesting the reevaluation of pre-staging of KI beyond the current 10-mile zone.

The NRC revised EP regulations to address requirements for KI; this revision became effective April 19, 2001. The revised rule requires that States with a population within the 10-mile EPZ of commercial nuclear power plants consider including KI as a protective measure for the general public to supplement sheltering and evacuation in the unlikely event of a severe nuclear power plant accident. The 2001 revision to the rule was based on early health effects data from the Chernobyl nuclear power plant accident, which showed an increase in thyroid cancer among children.

At that time, the Commission believed the final rule, together with the Commission's decision to provide funding for the purchase of a State's supply of KI, provided a proper balance between encouraging (but not requiring) the offsite authorities to take advantage of the benefits of KI and acknowledging the offsite authorities' role in such matters. By requiring consideration of the use of KI, the Commission recognized the important role of States and local governments in matters of emergency planning.

Section 127 of the Public Health Security and Bioterrorism Preparedness and Response Act of 2002 (the Bioterrorism Act) requires State and local governments to distribute, through the national KI stockpile, KI tablets to populations within 20 miles of a nuclear power plant. In a January 22, 2008 memo, Dr. Marburger, Director of the Office of Science and Technology Policy at the White House, announced his decision to invoke the waiver in Section 127(f) of the Bioterrorism Act, thus keeping the current 10-mile KI distribution zone. The rationale for waiving this requirement is that a more effective preventive measure currently exists for the extended zone covered by the Bioterrorism Act - namely, avoidance of exposure altogether through evacuation of the potentially affected population and interdiction of contaminated food. Analysis of radiological release events that could lead to adverse thyroid conditions beyond the current 10-mile zone shows that limiting or avoiding exposure to radiation through these mechanisms is practical and much more effective than the administration of KI in the proposed extended zone.

Based on the recommendation of the Potassium Iodide Working Group (which is composed of the Food and Drug Administration (FDA) Center for Drug Evaluation and Research, FDA's Center for Devices and Radiological Health, and National Institutes of Health), the FDA set the PAG of 5 rem projected dose to the child thyroid for the administration of KI. The staff has concluded that, based on currently available data, it is unlikely that the FDA thyroid dose PAGs were exceeded beyond 10 miles as a result of the accident at Fukushima. The staff will continue to monitor and evaluate the results of the findings as studies are conducted in and around the Fukushima site.

<u>Dependencies</u>

The staff has not identified any dependencies on other NTTF Recommendations.

<u>Staff Plan</u>

The issue of whether KI should be distributed beyond the 10-mile EPZ will be evaluated within ongoing efforts to address issues surrounding the use of KI. The staff plans to review information obtained from studies proposed by the Japanese Government and will propose any changes to policy regarding KI.

<u>Schedule and Milestones</u>

Using existing resources, the staff will continue to monitor and evaluate the population health studies proposed by the Japanese Government. This project will take 5 – 7 years to obtain meaningful data.

<u>Resources</u>

Office	FY 2013		FY 2014		estimated FY 2015 and beyond	
	FTE	Dollars, $K	FTE	Dollars, $K	FTE	Dollars, $K
NSIR	[ ]					[ ]
TOTAL	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

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Tier 3 – SECY-11-0137 Additional Recommendation 5

Program Plan for Transfer of Spent Fuel to Dry Cask Storage

Purpose

Following the event at Fukushima, several stakeholders submitted comments to the Commission and staff requesting that regulatory action be taken to require the expedited[3] transfer of spent fuel to dry cask storage. Based on past studies, the NRC has concluded that both SFPs and dry casks provide adequate protection of public health and safety and the environment, and that the likelihood of an accident involving a radiological release from the spent fuel pool remains extremely small. While the staff has concluded that public health and safety is adequately protected, the staff has determined that it should confirm, using insights from Fukushima, that both SFPs and dry cask storage continue to provide adequate protection, and assess whether any significant safety benefits (or detriments) would occur from expedited transfer of spent fuel to dry casks.

NTTF Recommendation and Other Direction

In SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, the staff identified a number of additional issues with a clear nexus to the Fukushima Dai-ichi event that may warrant regulatory action, but which were not included in the Near Term Task Force (NTTF) recommendations. One of the additional recommendations that warranted further consideration and potential prioritization was the evaluation of expedited transfer of spent fuel to dry cask storage.

Requirements

- GDC 61, "Fuel Storage and Handling and Radioactivity Control" of Appendix A to 10 CFR Part 50 requires that fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

- 10 CFR 50.54(hh)(2) requires licensees to develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling capabilities under the circumstances associated with loss of large areas of the plant as a result of explosions or fire.

[3] For the purposes of this program plan, expedited transfer is defined as the movement of spent fuel (stored in SFPs for more than five years) into dry cask storage earlier than is currently being conducted.

- Order EA-12-049 requires that licensees develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities following a beyond design basis external event.

- Order EA-12-051 requires that licensees install reliable means of remotely monitoring wide-range SFP levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event.

- 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Waste, and Reactor-Related Greater than Class C Waste,” establishes safety and physical protection criteria for storing spent nuclear fuel under a general license at a reactor site or under a site-specfic license at either a reactor site on away-from-reactor site.

Staff Assessment and Basis for Prioritization

Over the years, the staff has conducted technical studies of the safety of SFPs and the storage of spent fuel in dry cask storage systems. These studies have concluded that the current approaches to storage of spent fuel maintain safety. These studies include the following:

- NUREG-1353, “Regulatory Analysis for the Resolution of Generic Issue 82, ‘Beyond Design Basis Accidents in Spent Fuel Pools’,” issued April 1989. This report presents the staff’s evaluation of additional protective measures for the safe storage of spent fuel in high-density storage racks and included a regulatory analysis of alternatives to enhance the safety of spent fuel during beyond-design-basis accidents. The analysis determined that the risk from the storage of spent nuclear fuel in SFPs was dominated by beyond-design-basis earthquakes, but the objectives of the Commission’s Safety Goal Policy Statement were satisfied with no action. Based on the evaluation, the staff concluded that no new regulatory requirements were warranted concerning the use of high-density storage racks.

- NUREG-1738, “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Plants,” issued February 2001. This report provided the results of the staff evaluation of the potential accident risk in SFPs at decommissioning plants in the United States. This study was prepared to provide a technical basis for potential rulemaking that would relax emergency planning, insurance, and security requirements for permanently shutdown nuclear power plants. The study included a conservative modeling of SFP accidents consistent with its purpose. The results demonstrated that the overall SFP accident risk was well below the quantitative health objectives derived from the Commission’s Safety Goal Policy Statement.

- NUREG-1864, “A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant,” issued March 2007. This study assesses the risks from damage to fuel within dry cask storage systems and shows that the resulting calculated risk from storage of spent nuclear fuel in a dry cask storage system is extremely small. The study assumes that operators load and handle the casks in accordance with procedures and neglects the effects of postulated accidents on the nuclear power plant structures and

systems, but a generic frequency of heavy load drops is considered for estimating the risk from these events from fuel within the cask system.

- NUREG/CR-7017, "Preliminary, Qualitative Human Reliability Analysis for Spent Fuel Handling," issued February 2012. This report presents a preliminary, qualitative human reliability analysis (HRA) to examine, in a generic manner, how human performance of dry cask storage operations could plausibly lead to radiological consequences that impact the public and the environment. The report develops scenarios involving the misloading of spent fuel in a cask and cask drops, as well as other human performance aspects.

- NUREG/CR-7016, "Human Reliability Analysis-Informed Insights on Cask Drops," issued February 2012. This report documents HRA-informed insights on cask drops that may be used as an initial technical basis for activities aimed at reducing the potential for cask drops. The study investigated what should be included in a qualitative HRA for spent fuel and cask-handling operations to understand the potential for cask drops, and began building a technical basis for potential improvements to dry cask storage operations procedures and practices to reduce the likelihood of cask drops resulting from unsafe human actions. The study provided human performance insights on how potential cask drops may occur and, how they may be avoided or how the consequences of unsafe actions may be mitigated.

Following the event at Fukushima, several stakeholders submitted comments to the Commission and staff requesting that regulatory action be taken to require the expedited transfer of spent fuel stored in SFPs for more than five years to dry cask storage. Some stakeholders have stated that SFPs were not originally designed to handle the current amount of spent fuel being stored in the SFPs. Stakeholders have stated that a reduced amount of spent fuel in the SFPs would provide more effective cooling of the remaining fuel and reduce the inventory of radionuclides (e.g. cesium-137) that could be released if the fuel overheated.

As directed by the Commission in SRM-SECY-12-0025, dated March 9, 2012, the staff has undertaken regulatory actions, which originated from the NTTF recommendations to enhance reactor and SFP safety. On March 12, 2012, the staff issued Order EA-12-051 which requires that licensees install reliable means of remotely monitoring wide-range SFP levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event. In addition, the staff issued Order EA-12-049 which requires that licensees develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities following a beyond design basis external event. Upon full implementation of these Orders at nuclear power plants, SFP safety will be significantly increased.

Based on past studies, the NRC has concluded that both SFPs and dry casks provide adequate protection of public health and safety and the environment, and that the likelihood of an accident involving a radiological release from the spent fuel pool remains extremely small. Regarding the event at Fukushima, the information that the staff has to date indicates no significant offsite radioactive release from spent fuel stored in the SFP or the dry casks occurred. In fact, the events at Fukushima Dai-ichi demonstrated that SFPs are highly robust structures and that the fuel remained adequately cooled through addition of water to make-up for boiling. As such, there is no new information presently available from the Fukushima event that would indicate a

safety concern to mandate expedited transfer of spent fuel from SFPs to dry casks. Nevertheless, the staff will continue to monitor ongoing international research and recovery efforts to understand all of the effects from the Fukushima event, including those effects associated with the hydrogen explosion in Unit 4, and will appropriately follow-up on any potential safety issues which are identified. In conclusion, the staff has determined that its prior adequate protection determination remains valid.

While the staff has concluded that public health and safety is adequately protected, the staff has determined that it should confirm, using insights from Fukushima, that both SFPs and dry cask storage continue to provide adequate protection, and assess whether any significant safety benefits (or detriments) would occur from expedited transfer of spent fuel to dry casks. This assessment will include development of a technical basis through additional research and evaluation of the merits of any proposed regulatory action in accordance with established Commission policy and regulations, including the Backfit rule.

As the staff believes that there is ample regulatory basis for the current agency position that spent fuel storage is safe, the staff plans to evaluate whether there would be a substantial increase in the overall protection of public health and safety from expedited transfer of spent fuel into dry cask storage with consideration of the events at Fukushima Daiichi, the regulatory actions on SFPs being taken in Tier 1 and Tier 2 activities, and any other research needs identified. The staff plans to document its evaluation with respect to the Commission Safety Goals. If any new information or gaps in understanding are identified that could change the staff's current understanding of spent fuel safety, the staff will recommend that additional research be undertaken that will address these gaps. The staff will also evaluate the issue within the context of the current regulatory framework. Once it has all of the information necessary to complete the evaluation within the current regulatory framework, the staff will engage with stakeholders, including the public, industry, and the Advisory Committee on Reactor Safeguards (ACRS) and ultimately propose to the Commission any recommendations for regulatory action.

As a first step, the Office of Nuclear Regulatory Research (RES) has undertaken a study to produce updated consequence estimates for scenarios of interest related to SFPs. This Spent Fuel Pool Scoping Study (SFPSS) will evaluate the consequences associated with a large seismic event and its impact on the SFP. Prior studies have concluded that a beyond-design-basis earthquake accident scenario is the principle contributor to SFP risk. The SFPSS considers ground motion associated with a rare, but credible seismic event and uses structural analysis methods to determine the potential damage states, including some damage states that affect SFP integrity, for a BWR with a Mark I containment design. The SFPSS will use detailed modeling of the event progression to determine the consequences of any resulting fission product release for various cases that reflect the effect of changes in configuration during the operating cycle, the loading of the SFP, and the deployment of mitigation capabilities. Details of the SFPSS can be found in numerous public presentations, such as the March 14, 2012, Regulatory Information Conference presentation (accessible at https://ric.nrc-gateway.gov/docs/abstracts/esmaili_h-helton_d_hv_W16.pdf) and the ACRS meeting on April 12, 2012 (accessible in ADAMS at Accession No. ML12115A085). Preliminary insights from the SFPSS have already identified other areas where additional research would inform the staff's decision-making process.

Recently, the staff has identified additional research to complement the SFPSS, including human reliability analysis of the likelihood of successful deployment of mitigation capability. The staff has initiated an evaluation of this likelihood of success for the beyond design basis earthquake considered in the SFPSS. This evaluation will first determine the additional structure, system, and component failures inside and outside of the nuclear power station caused by that earthquake that were not needed for evaluating the accident progression for a single SFP in isolation but are needed for performing a mitigation reliability analysis. The success probabilities of performing the mitigative actions will then be evaluated.

The staff may conduct additional research to understand other considerations associated with the expedited transfer of spent fuel to dry cask storage. These other considerations include the operational and radiological risks associated with expedited transfer of spent fuel, financial costs, industry capabilities, and impact on spent fuel final disposal. Some of these issues have been the subject of recent industry studies performed by the Electric Power Research Institute (EPRI) (report located at http://www.epri.com under report no. EPRI-TR-1021049) and NAC International (report accessible in ADAMS at Accession No. ML11228A013), and may be considered in future agency activities. The EPRI study provided a preliminary estimate of the operational risks and radiological doses posed to workers, the financial aspects of requiring more casks, cask vendor capabilities to deliver the needed number of casks, and utility capacity to handle augmented loading campaigns.

In addition to examining the issues raised in the EPRI report, staff should also consider the impacts of the expedited transfer of spent fuel into dry storage on the Department of Energy's (DOEs) cask standardization program and final disposal. Should expedited transfer be required, it is expected that utilities would employ large capacity storage casks to minimize costs and handling. None of the proposed DOE repository designs were planned to accommodate the direct emplacement of large casks. Thus, the use of large canisters for storage may prove incompatible with a future repository design. The staff is also engaged in a significant effort with DOE and industry to address technical issues related to long term aging issues, such as canister and fuel cladding degradation. This DOE research effort could provide valuable insights that should be incorporated into the staff's assessment of the potential safety benefits of expedited spent fuel transfer to dry cask storage.

Consistent with guidance provided in NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," the staff will develop a regulatory analysis for any potential new or revised regulatory actions. The regulatory analysis provides a listing of the costs and benefits of a proposed action and is meant to serve as a decision tool for decisionmakers. When fully completed, these three analyses (SFPSS, HRA, other considerations) are collectively intended to help the staff determine what will be the necessary and sufficient inputs needed for conducting a regulatory analysis to address this issue.

Dependencies

The staff has determined that there are dependencies between the evaluation of this issue and Order EA-12-051 for all power reactor licensees to install reliable SFP instrumentation, and Order EA-12-049 requiring mitigation strategies for beyond-design-basis external events, including mitigation capabilities for SFPs. The full implementation of these orders is expected to enhance the capability to mitigate beyond design basis events that affect the ability to

continuously cool spent fuel. Addtionally, activities related to the Tier 2 item on SFP make-up capability could further enhance the capability to cool spent fuel.

Staff Plan

The staff plans to undertake the following activities:

- Complete research including the SFPSS, human reliability analysis for mitigation, and effects associated with increased fuel handling activities associated with loading and movement of casks to enhance its understanding of spent fuel safety.

- Evaluate the research stated above and other lessons learned from Tier 1 and 2 items on the staff's current understanding of spent fuel safety in accordance with the Commission Safety Goals and existing NRC studies and analyses.

- Gather stakeholder input, including from both the ACRS and public, on the results of its research and any potential regulatory actions.

- Develop a regulatory analysis for any potential new or revised regulatory action.

- Provide a recommendation on the need for any regulatory action to the Commission.

Schedules and Milestones

The staff does not currently have sufficient information regarding the scope of all of the additional research needed to develop a detailed schedule with milestones. However, the staff believes it can complete the scope of activities described in this plan within 5 years. More details on the staff's schedule will be provided in the next 6-month status update.

Resources

Office	FY 2012		FY 2013		FY 2014	
	FTE	Dollars, $K	FTE	Dollars, $K	FTE	Dollars, $K
NMSS	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
NRO	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
NRR	[ ]					[ ]
RES	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
Region I	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
TOTAL	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

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Tier 3 – SECY-12-0025 ACRS Additional Recommendation

Program Plan for Enhanced Reactor and Containment Instrumentation Withstanding Beyond-Design-Basis Conditions

Purpose

The purpose of this project plan is to define the short-term and long-term activities to support the identification of needs for enhanced reactor and containment instrumentation to withstand beyond-design-basis accident conditions. The plan also defines activities to support decisions on the need for any subsequent regulatory changes.

NTTF Recommendation and Other Direction

The staff accepted ACRS Recommendation 2(e) as a Tier 3 item.

- ACRS Recommendation 2(e) – "Selected reactor and containment instrumentation should be enhanced to withstand beyond-design-basis accident conditions."

- ACRS recommended adding this item to those in the NTTF report during its review at the Committee's 587th meeting on October 6, 2011. Subsequently, during its review of the prioritization of NTTF items at its 588th meeting on November 3-5, 2011, ACRS recommended that this be prioritized as Tier 2.

- As described in SECY-12-0025, Enclosure 3, the staff evaluated this issue and, based on the priority definitions of Tier 3 from SECY-11-0137, recommended that this activity be prioritized as Tier 3 because it requires further staff study and is dependent on other NTTF shorter term activities.

Regulations and Guidance

- GDC 13, "Instrumentation and Control," of Appendix A to 10 CFR Part 50 requires operating reactor licensees to provide instrumentation to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety.

- GDC 19, "Control Room," of Appendix A to 10 CFR Part 50 requires operating reactor licensees to provide a control room from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including LOCAs. In addition, operating reactor licensees must provide equipment (including the necessary instrumentation), at appropriate locations outside the control room, with a design capability for prompt hot shutdown of the reactor.

- GDC 64, "Monitoring Radioactivity Releases," of Appendix A to 10 CFR Part 50 requires operating reactor licensees to provide the means for monitoring the reactor containment atmosphere, spaces containing components to recirculate LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released as a result of postulated accidents.

- 10 CFR 50.34(f)(2)(ix)(c), requires that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after exposure to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100-percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

- 10 CFR 50.34(f)(2)(xvii) requires that licensees provide instrumentation to measure, record, and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points.

- 10 CFR 50.34(f)(2)(xix), requires operating reactor licensees to provide adequate instrumentation for use in monitoring plant conditions following an accident that includes core damage.

- RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," describes a method that the NRC staff considers acceptable for use in complying with the agency's regulations with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants. Regulatory Position 3 of Revision 4 of this RG provides a framework for identifying requirements for accident monitoring instrumentation that provides for expanded ranges for those variables that provide the most direct indication of the integrity of the three fission product barriers and provide the capability for monitoring beyond the normal range. Specifically, Revision 4 endorses the use of Institute for Electrical and Electronic Engineers (IEEE) Standard 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." Clause 5.1 of the IEEE Standard states that the range for Type C variables shall encompass, with margin, those limits that would indicate a breach in a fission product barrier. Further, Regulatory Position 4 of the RG states that Clause 4.1 of the IEEE Standard should be modified to state that Type A variables include those variables that are associated with contingency actions that are within the plant licensing basis and may be identified in written procedures. Clause 3.6 of the IEEE Standard defines contingency actions as "alternative actions taken to address unexpected responses of the plant *or conditions beyond its licensing basis*." (emphasis added).

- 10 CFR 52.47(a)(23) and 10 CFR 52.79(a)(38), require, for LWR designs, a description of design features for the prevention and mitigation of severe accidents.

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

During its review of the NTTF recommendations in SECY-11-0124 and SECY-11-0137, ACRS noted that Section 4.2 of the NTTF Report discusses how the Fukushima Dai-ichi operators faced significant challenges in understanding the condition of the reactors, containments, and SFPs because instrumentation was either lacking or giving erroneous readings. ACRS also noted that the Japanese Government has included the need for enhanced reactor and containment instrumentation in its list of key actions that will be taken in response to the accident at Fukushima.

ACRS specifically noted that past research efforts in this area had demonstrated systematic methodologies that could support the identification of plant information needs as well as location, range and ability to withstand environmental conditions for instrumentation systems for the risk-dominant severe accident sequences.

The staff agreed with the ACRS concern that adequate reactor and containment instrumentation is needed to support severe accident management. As a result of the post-TMI actions, requirements were put in place to enhance accident monitoring (the guidance for implementing these requirements is found in RG 1.97 Revision 2*, issued December 1980, and Revision 3, issued May 1983. However, the requirements were initially for design-basis events, not beyond-design-basis events. Current reactor and containment instrumentation is not specifically designed to remain functional under severe accident conditions. The current regulation in this area is 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," requirements for new reactor design certification and combined license applications to perform severe accident performance assessments that provide assessments of severe accident equipment needs, predicted environments and equipment survivability. For the instrumentation to provide information necessary to support operators in responding to severe accident events, the instrumentation must survive severe conditions and be provided with a functional supply of power. Revision 4 of RG 1.97, issued June 2006, states that licensees should provide instrumentation with expanded ranges and capable of surviving the accident environment (with a source term that considers a damaged core) in which it is located for the length of time its function is required.

The selection and survivability of instruments for the beyond design basis and severe accident environmental conditions will require additional study to address the ACRS concerns and formulate a recommendation for further regulatory actions. Key questions that require further study include the following:

- Is the current instrumentation identified in RG 1.97 adequate to cover the full range of beyond-design-basis conditions suggested by the Fukushima event?

- Will the instrumentation qualified to address the guidance of RG 1.97 survive with adequate capability to ensure monitoring of severe accident conditions?

The staff identified a number of NTTF Tier 1 actions that would affect the identification of needs for enhanced reactor and containment instrumentation: 1) NTTF Recommendation 2.3 actions to identify and address plant-specific vulnerabilities and verify the adequacy of monitoring and maintenance for protection features; 2) NTTF Recommendation 8 which will require strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and extensive damage mitigation guidelines (EDMGs); 3) NTTF Recommendation 4.1 which will require strengthening SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events; 4) Order EA-12-049, concerning mitigation strategies, and 5) Order EA-12-051, addressing SFP monitoring. Instrumentation needs and survivability should be considered as part of these higher level NTTF actions.

The staff determined that ACRS Recommendation 2(e) may improve safety, provided that applicable NTTF actions and further study identify significant improvements in reactor and containment instrumentation. Sufficient resource flexibility, including availability of critical skill

sets, exists; however, the staff prioritized this action as a Tier 3 recommendation because it will require further study and collaboration with higher tier recommendations.

Dependencies

- NTTF Recommendation 2.3 – This recommendation involves severe storm, seismic and flood protection walkdowns to identify and address plant-specific vulnerabilities and verify the adequacy of monitoring and maintenance for protection features. Resolution of this recommendation may reveal severe hazard conditions that may inform assessments of equipment survivability for severe accident instrumentation.

- NTTF Recommendation 4.1 – This recommendation involves strengthening SBO mitigation. Resolution of this recommendation will improve capabilities for powering equipment supporting core cooling and SFP cooling, as well as reactor coolant system and containment integrity in extended loss of ac conditions. These capabilities will inform assessments of equipment survivability for severe accident instrumentation.

- Order EA-12-049 – This order involves developing strategies to mitigate beyond-design-basis external events. These strategies will address both multiunit events and reasonable protection of equipment identified under such strategies. These capabilities will inform assessments of equipment survivability for severe accident instrumentation.

- Order EA-12-051 – This order involves installing enhanced SFP instrumentation to withstand beyond-design-basis external events to provide emergency responders with reliable information on the condition of the SFP. This will expand the list of instrumentation needed to fully monitor severe accident conditions and it will also inform assessments of equipment survivability.

- NTTF Recommendation 8 – This recommendation involves strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs, and will reveal site response needs for condition monitoring and instrumentation, which will support identification of severe accident instrumentation.

Staff Plan

The staff plans to undertake short-term and long-term regulatory activities to achieve the following:

(1) Ensure that licensees are appropriately considering instrumentation needs during implementation of actions for NTTF Recommendations 2.3, 4.1, and 8, and Order EA-12-049 and EA-12-051. The staff will engage stakeholders and monitor licensee activities in response to these actions to identify the applicable information needs from site-specific actions for severe accident scenario analysis and mitigation, extended loss of ac power capabilities, SFP instrumentation improvements, and integration of EOPs, SAMGs, and EDMGs.

(2) Obtain and review information from previous and ongoing research efforts for severe accident management analysis. In addition, coordinate with international and domestic

entities such as the International Atomic Energy Agency and DOE, as appropriate, to identify requirements for enhanced reactor, containment and other instrumentation to support severe accident management and gather information on accident instrumentation performance during the Fukushima event. This information will inform the assessment of, (a) additional information needed during beyond-design-basis accidents, (b) what constitutes survivable reactor, containment, and SFP instrumentation during beyond-design-basis accidents, (c) the conditions that the instrumentation must withstand to fulfill its intended function, and (d) the locations where such indications are needed (e.g., control room and/or remote location).

(3) Evaluate results from higher tier NTTF activities in coordination with the information obtained from applicable research efforts (international and domestic) to determine the need for any additional regulatory framework changes to address reactor, containment, and SFP instrumentation requirements to withstand beyond-design-basis accident conditions.

<u>Schedule and Milestones</u>

(1) Coordinate with staff addressing NTTF Recommendations 2.3, 4.1, and 8, and Orders EA-12-049 and EA-12-051 to ensure that instrumentation needs are considered. This will be done in accordance with the Tier 1 schedules.

(2) Obtain and review information and insights from previous and ongoing research and coordinate with international and domestic efforts to identify enhanced instrumentation needs. This will be completed in the next 2-3 years.

(3) Evaluate Tier 1 action information and information obtained from internal, domestic and international research to recommend regulatory framework changes, if any, for enhanced reactor and containment instrumentation. The schedule for this activity is to be determined.

<u>Resources</u>

Office	FY 2013		FY 2014		estimated FY 2015 and beyond	
	FTE	Dollars, $K	FTE	Dollars, $K	FTE	Dollars, $K
RES	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
NRR	[ ]					[ ]
NRO	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
TOTAL	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

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