Closure of Tier 3 Additional Recommendation

Enhanced Reactor and Containment Instrumentation for Beyond-Design-Basis Conditions

In SECY-15-0137, "Proposed Plans for Resolving Open Fukushima Tier 2 and 3 Recommendations," the U. S. Nuclear Regulatory Commission (NRC) staff provided the Commission an initial assessment of a Fukushima Dai-ichi lessons-learned recommendation from the Advisory Committee on Reactor Safeguards (ACRS) associated with enhancements to reactor and containment instrumentation to withstand beyond-design-basis accident conditions. In SECY-15-0137, the staff stated that additional interactions with the ACRS and external stakeholders were planned and more detailed documentation, incorporating insights from these interactions, would be provided to the Commission.

This final evaluation addresses the observations provided by the ACRS in their letter dated November 16, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15320A074), on the staff's initial assessments in SECY-15-0137. The NRC staff met publicly on January 7, 2016, with external stakeholders to gather information and insights regarding the issues and initial conclusions discussed in Enclosure 5 to SECY-15-0137, which provides the NRC staff's initial assessment of enhanced reactor and containment instrumentation for beyond-design-basis conditions. NRC staff also met with the ACRS Fukushima Subcommittee on February 18, 2016, and the ACRS Full Committee on March 3, 2016. The staff addresses insights from those interactions in the following justification for the closure of this recommendation. For the sake of completeness, this discussion includes the initial assessment from Enclosure 4 to SECY-15-0137 and the marked text provides additional discussions to address questions and insights from interactions with various stakeholders.

Background

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

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

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

As a result of the TMI accident, a set of generic safety issues was identified, including TMI Action Plan Item II.F.3, "Instruments for Monitoring Accident Conditions." The resolution of this item can be found in NUREG-0933, "Resolution of Generic Safety Issues," at http://nureg.nrc.gov/sr0933. TMI Action Plan Item II.F.3 addressed several concerns regarding the availability and adequacy of instrumentation to monitor plant variables and systems during and following an accident. This item was resolved by establishing new requirements as described in a December 17, 1982, letter to all licensees of operating reactors, applicants for operating reactors, and holders of construction permits (ADAMS Accession No. ML031080548). Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 2, included an expanded list of parameters and instrument ranges for licensees to consider when demonstrating that they met the underlying NRC requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria": General Design Criterion (GDC) 13, "Instrumentation and Control"; GDC 19, "Control Room"; and GDC 64, "Monitoring Radioactivity Releases."

Revision 2 to RG 1.97 provided instrumentation-related guidance to licensees for a variety of reactor accidents, but did not specifically call for devices to be designed and tested against the expected range of severe accident conditions. Revision 4 to RG 1.97 was developed, in part, to support the licensing of new reactors under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." As described later in this enclosure, new reactor applicants provide severe accident performance assessments that address severe accident equipment needs, predicted environments, and equipment survivability. Revision 4 to the guidance also states that new reactors should have instrumentation with expanded ranges and that are 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. Revision 4 of RG 1.97 acknowledges that current operating reactors are unaffected by the revision and previous revisions of the guidance remain in effect for those plants. The enhanced expectations for new reactors reflect Commission policies related to severe accidents and advanced reactor designs.

The industry developed severe accident management guidelines (SAMGs) in the 1990s to provide strategies and guidelines to mitigate the consequences of a severe accident. SECY-15-0065, "Proposed Rulemaking: Mitigation of Beyond-Design-Basis Events (RIN 3150-AJ49)," issued on April 30, 2015 (ADAMS Accession No. ML15049A201), discusses the history of the development of SAMGs. Enclosure 3 to SECY-15-0065, Section A.2, "Backfit Analysis of Rule Provisions that Constitute Backfits," discusses how the Commission's 1985 Severe Accident Policy Statement (50 FR 32138) led to SAMGs being implemented at licensee facilities on a voluntary basis by the end of 1998. If it is determined that adequate core cooling is no longer assured, the licensee exits the plant emergency operating procedures (EOPs) or other governing processes and enters the SAMGs. The SAMGs are symptom-based, pre-planned accident mitigation strategies that were developed using modern thermal-hydraulic and accident progression and consequence modeling. The SAMGs were developed for use in specific reactor designs and then adapted by individual licensees to reflect plant-specific design features and capabilities.

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

As part of the mitigation of beyond-design-basis events (MBDBE) rulemaking process, the staff proposes to endorse the use of the implementing guidance in the Nuclear Energy Institute (NEI) 14-01, "Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents." The guidance in NEI 14-01 supersedes guidance within NEI 91-04, "Severe Accident Closure Guidelines," and does not specifically state that an analysis be considered for the availability and survivability of instrumentation during severe accidents, as did the NEI 91-04 guidance. The newer guidance addresses the use of any available indications, including consideration of potential uncertainties, and the use of computational aids when direct diagnosis of key plant conditions cannot be determined solely from instrumentation.

Current Status

The NRC staff has completed its assessment of this recommendation. As discussed below, the staff has determined that the results of additional studies are unlikely to support new regulatory requirements related to enhanced reactor and containment instrumentation for beyond-design-basis conditions that would be warranted when evaluated against 10 CFR 50.109, "Backfitting," criteria for operating reactors or the issue finality provisions of 10 CFR Part 52 for new reactors. Although not needed to resolve the post-Fukushima Tier 3 item, the staff plans to continue participating in standard development organizations and updating regulatory guidance documents on the subject as part of its routine activities. For example, based on efforts by the Institute of Electrical and Electronics Engineers (IEEE) to provide guidance to address enhanced reactor and containment instrumentation for beyond-design-basis conditions[1], the NRC staff plans to update and provide guidance in RG 1.97 for such instrumentation. If licensees of currently operating reactors so choose, they can use the revised guidance found in the future revision of RG 1.97 to enhance their reactor and containment instrumentation on a voluntary basis.

The staff's initial assessment, found in SECY-15-0137, has been updated to reflect interactions with the Commission, ACRS, and other stakeholders. NRC staff met publicly on January 7, 2016, with external stakeholders to gather information and insights regarding this issue and initial conclusions discussed in Enclosure 5 to SECY-15-0137 dealing with enhancements to instrumentation. The staff also benefited from the observations provided by the ACRS in the letter dated November 16, 2015. NRC staff met with the ACRS Fukushima Subcommittee on February 18, 2016, and the ACRS Full Committee on March 3, 2016. The staff addresses insights from those interactions in the following justification for the closure of this recommendation. The following discussion includes and expands upon the initial assessment from Enclosure 5 to SECY-15-0137.

Discussion

The NRC staff's assessment of this recommendation considered recent Commission decisions that directly or indirectly affected the NRC staff's evaluation, including the work performed to support the MBDBE rulemaking and the work associated with Fukushima-related orders. The assessment also considers Commission decisions associated with the review of new reactor severe accident instrumentation issues. The staff's assessment includes an evaluation of the regulatory basis for enhanced capabilities for severe accident instrumentation, considering a review of instrumentation needs for implementing specific Commission-directed actions, previous and ongoing research efforts associated with severe accidents, and whether or not requirements to upgrade some instrumentation for operating reactors to withstand beyond-design-basis environments are warranted.[2]

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

[2] Discussions throughout this enclosure related to assessing possible regulatory actions refer to comparing the potential safety benefits of plant modifications against the thresholds defined in NRC guidance for the backfit rule. This guidance states that uncertainties and limitations should be addressed qualitatively and, to the extent practical, quantitatively in the supporting documentation for the proposed regulatory action. The SRM for SECY-15-0065 provides the Commission's decision that licensees' regulatory commitments to implement and maintain SAMGs are sufficient and no additional regulatory requirements are needed in this area.

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

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

Operating Reactors

Mitigation of Beyond-Design-Basis Events Rulemaking

The MBDBE rulemaking will, in part, make the requirements of Orders EA-12-049 and EA-12-051 generically applicable. These orders, as discussed later, include requirements associated with instrumentation.

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

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

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

The NRC staff has begun the process of identifying changes to the ROP to ensure appropriate oversight of SAMG activities as directed by the Commission in the SRM dated August 27, 2015. In an October 26, 2015, letter (ADAMS Accession No. ML15335A442), NEI informed NRC staff that each licensee will commit in writing by December 31, 2015, to perform timely updates of its site-specific SAMGs based on the revisions to Owners Group generic severe accident technical guidelines. The NEI informed the staff that licensees will also commit to ensuring that SAMGs are considered within the plant configuration management processes, integrated with other emergency response guideline sets and symptom-based emergency operating procedures, and validated. The staff has verified that all operating reactor licensees have submitted the aforementioned regulatory commitments in letters to the NRC.[3]

In a letter to NEI dated February 23, 2016 (ADAMS Accession No. ML16032A029), the NRC staff outlined its approach for developing the guidance to implement the Commission direction related to oversight of SAMGs. The NRC staff informed NEI that it would be pursuing a phased approach for updating the ROP relative to SAMGs. This phased approach is based on: (1) licensees' near-term commitments to make changes to plants' configuration management processes to ensure that the SAMGs reflect changes to the facilities over time, and (2) the longer-term commitment to upgrade site-specific SAMGs to ensure that they are consistent with the latest revision to generic SAMG guidance documents. The NRC staff will continue its interactions with external stakeholders and will incorporate appropriate changes to the inspection procedures and other guidance documents to ensure the regulatory commitments related to SAMGs are implemented and maintained.

The NRC staff's interactions with the ACRS and external stakeholders included a discussion of the Commission's decision to continue the existing regulatory treatment of SAMGs as an industry initiative. As discussed in SECY-15-0065, the imposition of regulatory requirements for the SAMGs and the related instrumentation are not justified based on quantitative measures. A similar basis and finding is presented in Enclosure 1 for possible enhancements to containments and venting systems that might be used during severe accident conditions. As reflected in the Commission's decision related to SAMGs, the NRC has regulations and policies that include thresholds for imposing additional regulatory requirements. These thresholds consider risk reduction and cost effectiveness in determining whether a potential enhancement should be imposed as a regulatory requirement. The Commission's recent decisions on SAMGs and other severe accident capabilities are in line with the conclusions in this assessment, that while such enhancements may be useful in responding to severe accidents, possible enhancements to reactor and containment instrumentation for beyond-design-basis

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[3] There have been seven combined licenses (COLs) issued in accordance with 10 CFR Part 52. The seven COLs are Vogtle Electric Generating Plant 3 and 4 (Vogtle), Virgil C. Summer Nuclear Station (Summer), Units 2 and 3, Fermi, Unit 3, and South Texas Project, Units 3 and 4. The NRC staff will work with COL holders and applicants to determine an appropriate mechanism to capture regulatory commitments related to the industry initiative on SAMGs.

conditions do not warrant regulatory actions when evaluated against established thresholds for operating reactors.[4] Severe accident instrumentation expectations for new reactors are discussed in the "New Reactors" section below.

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

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

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

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

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

[4] It is noted that studies by various organizations, including the National Academies of Science, the International Atomic Energy Agency (IAEA), and the Organization for Economic Cooperation and Development Committee on Nuclear Regulatory Activities, include recommendations in the aftermath of the Fukushima accident for possible enhancements to reactor and containment instrumentation as part of improvements to capabilities for dealing with beyond-design-basis events. These recommendations did not consider the cost effectiveness of such plant modifications in accordance with regulations and guidance applicable to the NRC.

Procedures and FLEX Support Guidelines or within the SAMGs. Typically, these parameters would include the following:[5]

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

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

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

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

Order EA-12-051 – Spent Fuel Pool Instrumentation

Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," dated March 12, 2012 (ADAMS Accession No. ML12056A044), requires nuclear power plants to install water level instrumentation in their spent fuel pools that must remotely report three distinct water levels: (1) normal level, (2) low level but still enough to shield workers above the pools from radiation, and (3) a level near the top of the spent fuel rods

[5] The staff notes that the guidance found in NEI 12-06 regarding mitigating strategies instrumentation capabilities does not include ensuring instrumentation such as containment hydrogen levels, containment radiation levels, core exit thermocouples, or radiation release levels, remain powered during an ELAP event. Rather, NEI 12-06 mitigating strategies instrumentation guidance is meant to provide another capability, beyond what existed before issuance of Order EA-12-049, to aid operators in determining whether or not decay heat is being removed under ELAP scenarios. Licensees may attempt to recover additional instrumentation referred to in SAMGs or use other assessment guidance described in this enclosure to ascertain the status of key safety functions during a severe accident.

where more water should be added without delay. Order EA-12-051 contains requirements regarding the instrumentation's ability to provide reliable readings at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period of time. Section 3.4 of NEI 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051," issued August 2012 (ADAMS Accession No. ML12240A307), provides expectations for the qualification of the instrumentation. The NRC staff endorsed the guidance found in NEI 12-02, with exceptions and clarifications, in JLD-ISG-12-03, "Compliance with Order EA-12-051 Reliable Spent Fuel Pool Instrumentation," dated August 29, 2012 (ADAMS Accession No. ML12221A339).

Order EA-13-109 – Containment Vent Order

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

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

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

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

New Reactors

The Commission's Severe Accident Policy Statement, issued in 1985 (50 FR 32138), documents the Commission's determination that for existing reactors, severe accidents must pose no undue risk to public health and safety. The Commission noted that this determination for existing reactors should not be viewed as implying that safety improvements in new plant designs should not be actively sought. The Commission further stated that it fully expects that vendors engaged in designing new plants will achieve a higher standard of severe accident safety performance than prior designs. This statement is consistent with the Commission's Policy Statement on Advanced Reactors, updated in 2008 (73 FR 60612), which states the Commission's expectation that advanced reactors will provide enhanced margins of safety. These policies led to the development of criteria for instrumentation enhancements for new reactors.

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

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

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

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

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

- Identify the high-level actions used to achieve a controlled, stable state.
- Define the accident time frames for each high-level action.
- Determine the equipment and instruments used to diagnose, perform, and verify high-level actions in each time frame.
- Determine the bounding environment within each time frame.
- Demonstrate reasonable assurance that the equipment will survive to perform its function within the severe environment.

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

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

1. Ensure that licensees and NRC staff are appropriately considering instrumentation needs when implementing site-specific actions (e.g., related to post-Fukushima regulatory actions).
2. Obtain and review information from previous and ongoing research efforts for severe accident management analysis. This task also involved coordination with international and domestic entities.
3. Evaluate the results of Tier 1 activities in coordination with the information obtained from applicable research efforts (international and domestic) to determine if possible requirements for enhanced instrumentation are warranted.

<u>Task 1</u>

The MBDBE rulemaking activities capture issues associated with NTTF Recommendations 4.1 and 8, and will make the requirements associated with Order EA-12-049 and EA-12-051 generically applicable. The guidance documents associated with the MBDBE rulemaking include: DG-1301, "Flexible Mitigation Strategies for Beyond-Design-Basis Events," issued April 2015; and DG-1317, "Wide-Range Spent Fuel Pool Level Instrumentation," and DG-1319, "Integrated Response Capabilities for Beyond-Design-Basis Events," both issued March 2015.

The staff considered whether additional instrumentation requirements were warranted during the development of the proposed MBDBE rule. The staff concluded that additional instrumentation was not needed for licensees to effectively implement SAMGS and that imposition of the requirement would not be in accordance with the backfitting requirements of

10 CFR 50.109. The Commission did not require SAMGs or additional instrumentation determining that licensees' voluntary measure to implement SAMGs was appropriate.[6]

As discussed above, the NRC staff also considered Tier 1 activities related to instrumentation requirements associated with Order EA-13-109. Guidance documents associated with this Order include expectations related to power availability and the environmental conditions expected with a loss of containment heat removal and an ELAP. The staff has confidence that the NRC and licensees are appropriately considering instrumentation needs when implementing site-specific actions for Tier 1 activities associated with the MBDBE rulemaking and Order EA-13-109.

Task 2

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

It should be noted that the NRC performs severe accident research in partnership with nuclear safety agencies and institutes in more than 20 countries. In addition, the NRC staff continues to engage the U.S. Department of Energy (DOE) and various trade organizations to ensure that Fukushima lessons-learned related to instrumentation capabilities during a severe accident are appropriately considered. Although the staff's assessment concludes that new regulatory requirements to enhance capabilities for severe accident instrumentation would not pass the backfitting analysis criteria for operating reactors, the NRC staff plans to continue to remain abreast of severe accident research activities with international and national organizations associated with the capabilities of instruments to withstand severe accident environments. A better understanding of instrumentation limitations in a severe accident environment has the potential to enhance an operator's ability to mitigate severe accidents and will help the NRC continuously verify the adequacy of its requirements.

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

1. IAEA

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

    - Draft IAEA TECDOC DD1135, "Assessment of Nuclear Power Plant Equipment Reliability Performance For Severe Accident Conditions"

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

[6] As previously mentioned, the staff notes that enhanced instrumentation could provide reliable and useful information to plant operators during some severe accident scenarios. However, the marginal safety benefit associated with the enhanced instrumentation does not warrant imposing new regulatory requirements. Licensees have developed technical support guidance for the SAMGs that include assessment techniques for plant parameter values and the status of plant safety functions during severe accident conditions.

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

- Report of the Committee on Nuclear Regulatory Activities (CNRA) Task Group on Accident Management, NEA/CNRA/R(2014)2, "Accident Management Insights after the Fukushima Daiichi NPP [Nuclear Power Plant] Accident"

2. Multinational Design Evaluation Program

- Evolutionary Pressurized Water Reactor Technical Experts Subgroup for Severe Accidents

3. EPRI

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

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

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

4. DOE

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

- Idaho National Laboratory report INL/EXT-13-28043, "TMI-2 - A Case Study for PWR [Pressurized-Water Reactor] Instrumentation Performance during a Severe Accident," March 2013

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

- Collaboration in a Japanese study on instrumentation performance at Fukushima

- Plant-specific studies on severe accident instrumentation needs and performance

- Report ORNL/TM-2015/278, "Post-Severe Accident Environmental Conditions for Essential Instrumentation for Boiling Water Reactors"

- Report INL/EXT-15-35940, "Scoping Study Investigating PWR Instrumentation during a Severe Accident Scenario"

5. National Academy of Sciences report

    - "Lessons Learned from the Fukushima Nuclear Accident for Improving Safety of U.S. Nuclear Plants," 2014

6. Interface with Standards Development Organization

    - The NRC staff plans to update RG 1.97 based on the planned new revision of IEEE Standard 497, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations"

<u>Task 3</u>

As part of Task 3, the NRC staff used the factors discussed above to determine if requirements for enhanced severe accident instrumentation could be justified under the NRC's regulatory framework. The NRC staff review considered the Commission's August 27, 2015, SRM that disapproved the imposition of a requirement for SAMGs. Although SAMGs are not a regulatory requirement, they are being voluntarily upgraded in response to Fukushima lessons learned, and the NRC staff is assessing these upgrades.[7] The NRC staff's evaluation also included a review of instrumentation relied upon for the MBDBE rulemaking and the instrumentation relied upon for SAMGs. Further, the NRC staff considered ongoing work on IEEE Standard 497. The discussion that follows provides the results of the staff's evaluation.

Operating Reactors

Using the insights described above, the staff has determined that there is little likelihood that further study or research would make it necessary to recommend additional requirements for licensees to enhance reactor and containment instrumentation to support monitoring capability during severe accidents. Based largely on the analyses completed for the MBDBE rulemaking, the staff has concluded that the imposition of such a regulatory requirement would not represent a substantial safety benefit to public health and safety. As a result, enhanced reactor and containment instrumentation requirements are unlikely to satisfy the criteria in 10 CFR 50.109 for backfitting an operating reactor. The NRC staff's determination is also based on consideration of the substantial safety improvements already being implemented as part of NRC's post-Fukushima regulatory actions, such as Order EA-12-049, Order EA-13-109, the MBDBE rulemaking, and voluntary industry initiatives.

Quantified Risk Information

Enclosure 3 of SECY-15-0065 (ADAMS Accession No. ML15049A212), Section A.2, "Backfit Analysis of Rule Provisions that Constitute Backfits," discusses quantified risk information as it relates to imposing SAMGs as a requirement in the MBDBE rulemaking. The document provides estimates of the risk of latent cancer fatality compared against the Commission's

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[7] The staff's evaluation did not include a detailed assessment of specific instrumentation qualifications and survivability during severe accident conditions. This limited evaluation reflects the Commission's decision provided in the SRM for SECY-15-0065 to accept regulatory commitments related to SAMGs and not to impose additional regulatory requirements in this area.

Safety Goal Policy quantitative health objective (QHO), which is a measure that equates to 1/10 of 1 percent of the individual latent cancer fatality risk. The quantitative metric for the individual latent cancer facility risk is approximately $2x10^{-6}$ per reactor year. The analysis concludes that SAMGs would have a small safety benefit and would not significantly enhance the margin to the QHO. The conclusion is based on the risk of a severe accident being low and that existing emergency preparedness requirements ensure that the surrounding population is adequately protected in the unlikely event a severe accident occurs. The staff notes that, given that SAMGs could not be justified based on quantified risk information, the imposition of enhanced reactor and containment instrumentation requirements to further improve SAMGs are similarly not justified based on risk.

The SAMGs were developed and implemented based on a philosophy that makes use of available instrumentation, includes backup or alternative means for determining plant conditions when the primary means become unavailable or unreliable, and includes a course of action to follow when the event degrades to the point where there is no reliable instrumentation available.

The NRC staff compared the approach described in IAEA Report NP-T-3.16 to U.S. approaches used as part of compliance with Order EA-12-049 and in the SAMGs. Specifically, Annex I to IAEA Report NP-T-3.16 includes a detailed discussion of the SA-Keisou approach being used in Japan (in English, SA-Keisou means severe accident instrumentation and monitoring systems). The NRC staff reviewed several licensees' mitigating strategies developed under Order EA-12-049 to identify the list of required instrumentation and compared those instruments with the SA-Keisou list. The staff found various differences between these lists, depending on the specific reactor type and plant. On the other hand, Boiling Water Reactor Owners Group (BWROG) and Pressurized Water Reactor Owners Group (PWROG) SAMG instrumentation is similar to the Japanese SA-Keisou instrumentation (with some exceptions). The exceptions are limited in nature (e.g., the SA-Keisou instrumentation includes BWR drywell water level and PWR reactor cavity level instrumentation that are not found in the SAMGs).

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

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

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

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

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

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

New Reactors

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

Regarding imposition of provisions in Order EA-12-049 related to providing power to instrumentation needed to implement MBDBE strategies should an ELAP occur, any backfitting and finality issues were addressed as part of the issuance of the orders. Therefore, for new reactors, in addition to the equipment survivability analysis discussed above, the staff notes that as a result of Order EA-12-049, strategies will be implemented to ensure mitigating strategies instrumentation will remain powered during an ELAP.[8]

Additional Discussions

In a letter to the Commission on SECY-15-0137, dated November 16, 2015 (ADAMS Accession No. ML15320A074), the ACRS stated:

> The staff plans to document the basis for closing this recommendation and interact with the ACRS and external stakeholders prior to reporting to the Commission by March 2016. Part of the basis for the staff's conclusion is that calculational aids could be used to supplement or replace data from instruments when required. We intend to explore with the staff and industry severe accident instrumentation availability, capability, and reliability, as well as the detailed SAMG diagnostic approaches. The staff should include a detailed demonstration of how the SAMGs and calculational aids are capable of leading the operators to take the correct actions, even if minimal instrumentation is available or their indications are suspect. Validation work used to determine what instrumentation is necessary before, during, and subsequent to a severe accident will also be examined.

In a January 7, 2016, public meeting (summary available in ADAMS at Accession No. ML16013A277), industry representatives provided a discussion of the guidelines used to validate instrument readings to determine emergency operating procedure and SAMG implementation. During the presentation, the industry representatives explained how the Owners Group generic SAMG guidance provides direction to plant operators on how to obtain

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8 There have been seven COLs issued in accordance with 10 CFR Part 52. Of these seven COLs, two (i.e., Vogtle 3 and 4) were issued prior to issuance of Order EA-12-049 on March 12, 2012. As a result, Order EA-12-049 was issued to Vogtle 3 and 4 and the licensee's obligation to meet the requirements in Order EA-12-49 are the same as operating reactors that received their licenses under 10 CFR Part 50. The requirements of Order EA-12-049 that are applicable to the five COLs that were issued after Order EA-12-049 (i.e., Summer, Units 2 and 3; Fermi, Unit 3; and South Texas Project, Units 3 and 4) were included as conditions of those licenses.

plant status information in extreme conditions. In addition to this guidance in the SAMGs, the industry representatives explained how the FLEX support guidance provides direction on taking manual readings from available instrumentation.

During the public meeting, the industry representatives described how the SAMGs include technical support guidance (TSGs) and how licensees would use that tool during a response to a severe accident. The TSGs describe actions the emergency response organization personnel may take to execute the mitigating actions developed from the emergency procedure guidelines (EPG) and severe accident guidelines (SAG). The TSGs include four interrelated assessments:

1. control parameter assessment guideline
2. plant status assessment guideline
3. function status assessment guideline
4. EPG/SAG action assessment guideline

Licensee personnel would use the control parameter assessment guideline to validate the instrument readings to determine EOP/SAMG control parameter values. This guideline helps emergency responders to:

- validate control parameter readings
- monitor and trend the important parameters for an event
- adapt frequency of monitoring to how fast the parameters are changing
- perform TSG calculations to support parameter assessment

Emergency responders use the following principles when validating instrumentation:

- Use other indications (indirect measures) to confirm readings, to ensure that assessments are confirmed from two or three sources.

- When actions are taken, monitor parameters to ensure that instruments respond as would be expected (e.g., reactor pressure vessel (RPV) level increases after the commencement of RPV vessel injection).

- Recognize that trends and changes in trends may be true even when absolute values or readings are not.

- Consider physical explanations for instrument readings. For example, actual RPV water level may be above or below the instrument taps for the level instrument which could lead to erroneous RPV level instrumentation readings.

During a closed session with the ACRS Fukushima Subcommittee on February 18, 2016, industry representatives provided a presentation to the subcommittee similar to presentations previously provided to the NRC staff. The presentation provided practical examples and insights into the use of SAMGs, the role of instrumentation, and the assessment guidance.

The role of calculational aids in assisting operators during a severe accident were also discussed during the ACRS Fukushima Subcommittee February 18, 2016, meeting. Some of the specific details associated with the calculational aids are proprietary, but the staff notes that the SAMG calculational aids are similar in concept to those found in Response Technical Manual (RTM) 96 (NUREG/BR-0150, Volume 1, Revision 4; ADAMS Accession No. ML091980341) that the staff would use to assess core damage during a severe accident. Section A of RTM-96 provides high-level calculational aids to be used to assess the condition of the core.

Example calculational aids include Table A-2 in RTM-96, which provides a sequence of core damage versus time that the reactor core is uncovered.[9]

Although NRC staff assessment and SAMG calculational aids both provide tools to assist the staff and operator in performing a reactor core damage assessment, the SAMGs also provide actions for licensees to consider based on the core damage assessment, including options for restoring core cooling. In extreme events, with minimal to no instrumentation inside containment available, the SAMGs direct licensees to consider pumping water into the reactor to restore core cooling or provide core debris cooling. Active debris cooling with containment flooding or spraying is addressed in SAMGs and represents an effective element of fission product release reductions.

The November 16, 2015, ACRS letter also noted that validation work used to determine what instrumentation is necessary before, during, and subsequent to a severe accident will also be examined by the ACRS. The NRC staff notes that SAMGs are based on the concept of using available resources (including instrumentation) to mitigate a severe accident, such that if a key instrument is not available for any reason, alternate instruments are used. The instrumentation that is available and that might be used before, during and after a severe accident are discussed in documents such as RG 1.97, licensing documents, SAMGs, and supporting technical guidance documents.

Summary of Staff's Assessment

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

For new reactors, the Commission policy decisions in the 1990s resulted in equipment survivability evaluations that have been, and will continue to be, performed to provide

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9 NRC staff continually assesses its incident response tools. As discussed in SECY-14-0027, "Review of Analysis Codes Used During the Fukushima Incident" (ADAMS Accession No. ML14016A478), a computer-based version of RTM-96 has been developed to assist in the technical analysis for determining core damage using the available input parameters. The Response Technical Tools package provides a dozen standalone assessment tools that allow analysts to predict, recognize, and initially estimate the degree of core damage using a limited set of parameters.

reasonable assurance that the equipment provided for severe accident protection will operate in the severe accident environment for which it is intended and over the time span for which it will be needed.

For both operating and new reactors, enhancements to the power supplies for mitigating strategies instrumentation have been, or will be, implemented in response to Order EA-12-049[10] and its associated rulemaking. This limited set of instrumentation provides the parameters necessary to demonstrate the success of the strategies at maintaining key safety functions, as well as indicating imminent or actual core damage to facilitate a decision to manage the response to the event within the emergency operating procedures and FLEX support guidelines or within the SAMGs. Providing additional power sources to this instrumentation throughout an accident's progression should aid licensee's understanding of the condition of the reactor vessel, containment, and SFPs prior to instrumentation becoming unavailable or unreliable under severe accident environmental conditions.

Stakeholder Interactions

Four public meetings associated with the development of the SECY-15-0137 recommendations were held. These meetings were:

- An October 6, 2015, public meeting with the ACRS Fukushima Subcommittee
- An October 20, 2015, Category 2 public meeting with the Fukushima steering committee
- A November 5, 2015, public meeting with the ACRS Full Committee
- A November 17, 2015, public meeting with the Commission

The staff has had other interactions beyond those above related to the enhanced instrumentation recommendation found in Enclosure 5 of SECY-15-0137. Specifically, a public meeting was held on January 7, 2016, during which the staff heard from representatives of the nuclear industry, nongovernment organizations, and members of the public. The staff considered insights from this meeting and the ACRS letter dated November 16, 2015, on SECY-15-0137 and prepared a "white paper" to support further interactions with the public and ACRS (ADAMS Accession No. ML16020A245). The staff's final evaluations benefited from interactions with the ACRS in February and March 2016, as discussed in the letter from ACRS dated March 15, 2016 (ADAMS Accession No. ML16075A330).

The comments received during the January 7, 2016, public meeting can be found in the meeting summary dated January 20, 2016 (ADAMS Accession No. ML16013A277). The NEI noted during the meeting that the industry agreed with the NRC staff's finding that further study is unlikely to identify a need for regulatory action related to enhanced instrumentation. The NEI's other comments related to enhanced instrumentation can be found in the "Discussion" section of this document. Paul Gunter of Beyond Nuclear questioned how the NRC was going to exercise oversight of SAMGs, noting previous cases where voluntary initiatives did not meet NRC expectations. As noted in the "Discussion" section above, the Commission provided direction to the NRC staff in an SRM dated August 27, 2015, to update the ROP to explicitly provide

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[10] As discussed above in footnote 8, the applicable portions of EA-12-049 for Summer, Units 2 and 3; Fermi, Unit 3; and South Texas Project, Units 3 and 4 are provided in a license condition to the 10 CFR Part 52 COLs.

periodic oversight of industry's implementation of SAMGs. The NRC staff is in the process of implementing this direction from the Commission. The NRC staff's efforts to provide a process in the ROP for oversight of SAMG implementation addresses the concern related to the historical lack of oversight of licensee's implementation and maintenance of SAMGs.

Conclusion and Recommendation

Based on the evaluation described above, the NRC staff concludes that further regulatory action on this recommendation is not warranted and it is closed.

Regarding the initiative to update RG 1.97, the update will include the revision to IEEE Standard 497 and will take approximately 1 year after the revision to IEEE Standard 497 is issued. As discussed above, the revision to IEEE Standard 497 is scheduled for completion in calendar year 2016. Licensees of currently operating reactors can use the guidance found in the revision of RG 1.97 to enhance their reactor and containment instrumentation on a voluntary basis. New reactor applicants will continue to assess equipment survivability for reactor and containment instrumentation for beyond-design-basis events, in accordance with Commission policy.