Proposed Resolution Plan for Tier 3 Additional Recommendation

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

Background

As directed by staff requirements memorandum (SRM) to SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated August 19, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112310021), the staff sought to identify additional recommendations related to lessons learned from the Fukushima Dai-ichi event, beyond those identified in the Near-Term Task Force (NTTF) report. Many additional recommendations were received from U.S. Nuclear Regulatory Commission (NRC) staff and external stakeholders, including the Office of Science and Technology Policy, Congress, international counterparts, other Federal and State agencies, nongovernmental organizations, the public, and the nuclear industry. These issues were raised in a variety of 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 Advisory Committee on Reactor Safeguards (ACRS) noted that Section 4.2 of the NTTF report discusses how the Fukushima operators faced significant challenges in understanding the condition of the reactors, containments, and spent fuel pools (SFPs) because the existing design-basis instrumentation was either lacking electrical power or was providing erroneous readings. As a result, an additional recommendation was developed to address the regulatory basis for requiring reactor and containment instrumentation to be enhanced to withstand beyond-design-basis accident conditions. This activity was prioritized as Tier 3 because it required further staff study and depended on the outcome of other lessons learned activities. The program plan for this recommendation was detailed in SECY-12-0095, "Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012 (ADAMS Accession No. ML12208A210).

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

As a result of the TMI accident, a set of generic safety issues was identified, including TMI Action Plan Item II.F.3, "Instruments for Monitoring Accident Conditions." The resolution of this item can be found in NUREG-0933, "Resolution of Generic Safety Issues," at http://nureg.nrc.gov/sr0933. 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" (GDC), GDC 13, "Instrumentation and Control;" GDC 19, "Control Room;" and GDC 64, "Monitoring Radioactivity Releases."

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

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

Current Status

The NRC staff has completed its assessment of this recommendation. As discussed below, the staff has determined that 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, "Licenses, Certifications, and Approvals for Nuclear Power Plants," 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, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," for such instrumentation. If licensees of currently operating reactors so choose, they can use the revised guidance found in the future revision of RG 1.97 to enhance their reactor and containment instrumentation on a voluntary basis.

Discussion

The NRC staff's assessment of this recommendation considered recent Commission decisions that directly or indirectly affected the NRC staff's evaluation, including the work performed to support the Mitigation of Beyond-Design-Basis Events (MBDBE) rulemaking and the work associated with Fukushima-related orders. The assessment also considers Commission decisions associated with the review of new reactor severe accident instrumentation issues. The staff's assessment 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.

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.

<u>Operating Reactors</u>

Mitigation of Beyond-Design-Basis Events Rulemaking

The Mitigation of Beyond-Design-Basis Events 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 <u>not</u> 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

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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.

> 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.

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

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

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

The minimum set of parameters necessary to support the FLEX strategy is discussed in Section 3.2.1.10 of Nuclear Energy Institute (NEI) 12-06, Revision 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," issued May 2012 (ADAMS Accession No.

ML12242A378). The NRC endorsed NEI 12-06, with clarification, in Japan Lessons-Learned Directorate (JLD) Interim Staff Guidance (ISG) JLD-ISG-2012-01, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated August 29, 2012 (ADAMS Accession No. ML12229A174). NEI 12-06, Section 3.2.1.10, states the following:

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

PWRs [Pressurized Water Reactors]	BWRs [Boiling Water Reactors]
• SG [Steam Generator] Level • 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 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.

<u>New Reactors</u>

The Commission's Severe Accident Policy Statement, issued in 1985 (50 FR 32138), documents the Commission's determination that for existing reactors, severe accidents must pose no undue risk to public health and safety. The Commission noted that this determination for existing reactors should not be viewed as implying that safety improvements in new plant designs should not be actively sought. The Commission further stated that it fully expects that vendors engaged in designing new plants will achieve a higher standard of severe accident safety performance than prior designs. This policy led to the development of criteria for instrumentation enhancements for new reactors.

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

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

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

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

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

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

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

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

- Determine the bounding environment within each time frame.

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

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

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

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

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

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

Task 1

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.

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 NRC staff has confidence that it and licensees are appropriately considering instrumentation needs when implementing site-specific actions for Tier 1 activities associated with the MBDBE rulemaking and Order EA-13-109.

Task 2

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

It should be noted that the NRC performs severe accident research in partnership with nuclear safety agencies and institutes in more than 20 countries. In addition, the NRC staff continues to engage the U.S. Department of Energy (DOE) and various trade organizations to ensure that Fukushima lessons-learned related to instrumentation capabilities during a severe accident are appropriately considered. Although the staff's assessment concludes that new regulatory requirements to enhance capabilities for severe accident instrumentation 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. International Atomic Energy Agency (IAEA)

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

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

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

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

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

3. Multinational Design Evaluation Program

    - Evolutionary Pressurized Water Reactor Technical Experts Subgroup for Severe Accidents

4. EPRI

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

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

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

5. DOE

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

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

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

    - Collaboration in a Japanese study on instrumentation performance at Fukushima

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

    - Draft report ORNL/TM-2015/278, “Post-Severe Accident Environmental Conditions for Essential Instrumentation for Boiling Water Reactors”

    - Draft report INL/EXT-15-35940, “Scoping Study Investigating PWR Instrumentation during a Severe Accident Scenario”

6. National Academy of Sciences (NAS) report

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

7. Interface with Standards Development Organization

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

<u>Task 3</u>

As part of Task 3, the NRC staff used the factors discussed above to determine if requirements for enhanced severe accident instrumentation could be justified under the NRC’s regulatory framework. The NRC staff review 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. The NRC staff’s evaluation also included a review of instrumentation relied upon for the MBDBE rulemaking and the instrumentation relied upon for SAMGs. Further, the NRC staff considered ongoing work on IEEE Standard 497. The discussion that follows provides the results of the staff’s evaluation.

Operating Reactors

Using the insights described above, the staff has determined that there is little likelihood that further study or research would enable the NRC to recommend additional requirements for licensees to enhance reactor and containment instrumentation to support monitoring capability during severe accidents. Based largely on the analyses completed for the MBDBE rulemaking, the staff has concluded that the imposition of such a regulatory requirement would not represent a substantial safety benefit to public health and safety. As a result, enhanced reactor and containment instrumentation requirements are unlikely to satisfy the criteria in 10 CFR 50.109 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 Safety Goal Policy quantitative health objective (QHO), which is a measure that equates to 1/10 of 1 percent of the individual latent cancer fatality risk. The quantitative metric for the individual latent cancer facility risk is approximately $2x10^{-6}$ per reactor year. The analysis concludes that SAMGs would have a small safety benefit and would not significantly help plants maintain margin to the QHO. The conclusion is based on the risk of a severe accident being low and that existing emergency preparedness requirements ensure that the surrounding population is adequately protected in the unlikely event a severe accident occurs. The staff notes that, given that SAMGs could not be justified based on quantified risk information alone, the imposition of enhanced reactor and containment instrumentation requirements to further improve SAMGs are similarly not justified based on risk.

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

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

The NRC staff closely reviewed the purpose, assumptions, approaches, and considerations for the SA-Keisou approach versus the approach used for Order EA-12-049 compliance. The SA-Keisou approach assumes worst-case, severe accident conditions, while the mitigating strategies approach assumes an ELAP and loss of normal access to the ultimate heat sink, but not core damage. The mitigating strategies parameters were selected to 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 parameters necessary to support strategy implementation.

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

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

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

As described above, the NRC staff believes that it is 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.

<u>Summary of Staff’s Assessment</u>

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

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

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

Stakeholder Interactions

The NRC staff provided the Fukushima subcommittee of the ACRS an overview of the staff's plans to resolving the open Tier 2 and 3 recommendations during a meeting held on October 6, 2015. A similar meeting is planned with the ACRS full committee on November 5, 2015. In addition, the staff provided an overview of its proposed resolution plans for all the open Tier 2 and 3 recommendations during a Category 2 public meeting held on October 20, 2015. The staff expects to conduct additional focused meetings on these recommendations with the ACRS and external stakeholders to support documenting its final analysis.

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

Conclusion and Recommendation

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

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

Resources

The staff estimates that approximately 1.0 full-time equivalent staff (FTE) is needed in fiscal year (FY) 2016 and 0.1 FTE in FY 2017 to complete the staff's assessment of IEEE Standard 497, update RG 1.97, support public interactions, and participate in ACRS meetings. The resources for FY 2016 are currently budgeted in the Operating Reactors Business Line, Licensing and Research Product Lines, Fukushima NTTF Product. If the staff identifies the need for additional resources in FY 2017 or beyond as it finalizes its evaluation, those resource needs will be addressed through the planning, budget, and performance management process.

Office	FY 2016	
	FTE	Dollars, $K
RES	0.8	-----
NRR	0.1	-----
NRO	0.1	-----
TOTAL	1.0	-----