[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

[Docket Nos. PRM-50-97 and PRM-50-98; NRC-2011-0189 and NRC-2014-0240]

RIN 3150-AJ49

Mitigation of Beyond-Design-Basis Events

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is proposing to amend its regulations that establish regulatory requirements for nuclear power reactor applicants and licensees to mitigate beyond-design-basis events. The NRC is proposing to make generically applicable requirements in Commission orders for mitigation of beyond-design-basis events and for reliable spent fuel pool instrumentation. The rule proposes to establish regulatory requirements for Severe Accident Management Guidelines (SAMGs) as part of an integrated response capability, including supporting requirements for command and control, drills, training and change control. The rule would set requirements for enhanced onsite emergency response capabilities and for mitigating strategies for new reactor designs. Finally, the proposed rule would address a number of petitions for rulemaking (PRMs) submitted in the aftermath of the March 2011 Fukushima Dai-ichi event. This rulemaking is applicable to power reactor licensees; power reactor construction permit, design certification, design approval, and license applicants; and decommissioning power reactor licensees. This rulemaking combines two NRC

activities for which documents have been published in the *Federal Register* - Onsite Emergency Response Capabilities (RIN 3150-AJ11; NRC-2012-0031) and Station Blackout Mitigation Strategies (RIN 3150-AJ08; NRC-2011-0299). The new identification numbers for this consolidated rule are RIN 3150-AJ49 and NRC-2014-0240.

DATES: Submit comments by **[INSERT DATE 75 DAYS FROM DATE OF PUBLICATION]**. Comments received after this date will be considered if it is practical to do so, but the Commission is able to ensure consideration only for comments received before this date. A public meeting will be held during the public comment period; refer to the NRC's public meeting schedule on the NRC Web site, www.nrc.gov or directly at http://meetings.nrc.gov/pmns/mtg.

ADDRESSES: You may submit comments by any of the following methods (unless this document describes a different method for submitting comments on a specific subject):

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2014-0240. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical questions contact the individuals listed in the FOR FURTHER INFORMATION CONTACT section of this document.
- **E-mail comments to:** Rulemaking.Comments@nrc.gov. If you do not receive an automatic e-mail reply confirming receipt, then contact us at 301-415-1677.
- **Fax comments to:** Secretary, U.S. Nuclear Regulatory Commission at 301-415-1101.
- **Mail comments to:** Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff.

- **Hand deliver comments to:** 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. (Eastern Time) Federal workdays; telephone: 301-415-1677.

For additional direction on obtaining information and submitting comments, see "Obtaining Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

FOR FURTHER INFORMATION CONTACT: Timothy Reed, Office of Nuclear Reactor Regulation, telephone: 301-415-1462, e-mail: Timothy.Reed@nrc.gov; Eric Bowman, Office of Nuclear Reactor Regulation, telephone: 301-415-2963, e-mail: Eric.Bowman@nrc.gov; U.S. Nuclear Regulatory Commission, Washington DC 20555-0001.

SUPPLEMENTARY INFORMATION:

EXECUTIVE SUMMARY:

A. Need for the Regulatory Action

The U.S. Nuclear Regulatory Commission (NRC) is proposing to amend its regulations to establish regulatory requirements for nuclear power reactor applicants and licensees to mitigate beyond-design-basis events. The NRC is proposing a rule that would make Commission Order EA-12-049 and Order EA-12-051 generically applicable, establish regulatory requirements for Severe Accident Management Guidelines (SAMGs) as part of an integrated response capability including supporting requirements for command and control, drills, training and change control, include requirements for enhanced onsite emergency response capabilities, provide requirements for mitigating strategies for new reactor designs, and address a number of petitions for rulemaking (PRMs) submitted in the aftermath of the March 2011 Fukushima Dai-ichi event. This rulemaking would be applicable to operating power reactor licensees; power reactor construction permit, design certification, design approval, and license applicants; and decommissioning power reactor licensees. The NRC is conducting this rulemaking to amend the regulations to reflect requirements imposed on current licensees by order and to reflect the lessons learned from the Fukushima accident.

B. Major Provisions

Major provisions of this proposed rule include amendments or additions to parts 50 and 52 of Title 10 of the *Code of Federal Regulations* (10 CFR) that would:

- Revise the 10 CFR parts 50 and 52 “Content of application” requirements to reflect the additional information that would be required for applications.
- Add proposed § 50.155, which contains beyond-design-basis mitigation requirements that would make Orders EA-12-049 and EA-12-051 generically applicable; requires an integrated response capability for beyond-design-basis events that includes the integration of three guideline sets with the existing emergency operating procedures; training requirements; drills or exercise requirements; and change control requirements, and establishes requirements that would apply only to new reactor designs.
- Revise 10 CFR part 50, appendix E to include enhanced capabilities for assessing the impact and release of radioactive materials for multi-unit events; to remove references to specific technology for each licensee’s emergency response data system; to include enhanced capabilities for onsite and offsite communications; and to add staffing analysis requirements to address multi-unit events.

C. Costs and Benefits

The NRC prepared a draft regulatory analysis to determine the expected quantitative costs and benefits of the proposed rule, as well as qualitative factors to be considered in the NRC's rulemaking decision. The draft analysis demonstrates that the proposed rule is justified. The draft analysis examines the benefits and costs of the proposed rule requirements relative to the baseline (i.e., no action alternative). Additionally, the draft analysis estimates the historical costs incurred as a result of implementation of Order EA-12-049, Order EA-12-051, and related industry initiatives. The proposed rule encompasses provisions that fall into two groups: 1) those within the scope set forth in Order EA-12-049 or Order EA-12-051, as well as related industry initiatives; and 2) those provisions associated with the new regulatory requirements for licensees to develop, implement, and maintain SAMGs, as well as the NRC's rulemaking-related costs. Because the NRC uses a no action baseline to estimate incremental costs, the total cost of the proposed rule largely results from imposition of SAMG-related requirements and excludes the costs that were incurred for implementation of the orders and related industry initiatives. As a result of the proposed rule, the NRC estimates that the industry as a whole would incur a total one-time cost of $30 million, followed by an annual cost of $2.4 million. The total present value of these costs is $58 million (using a 7 percent discount rate) and $72 million (using a 3 percent discount rate) over a 63-year period. The average power reactor site would incur a one-time cost of approximately $510,000, followed by an annual cost of approximately $42,000.

The proposed rule would result in a total one-time cost to the NRC of $1.1 million to complete the rulemaking (i.e., complete the proposed rule, analyze public comments, hold public meeting(s), and develop the final rule and regulatory guidance) and to oversee implementation of the SAMG-related requirements. This one-time cost would be followed by an annual cost of approximately $170,000 for SAMG-related requirements.

The draft regulatory analysis concludes that the costs of this proposed rule are justified in view of the quantitative and qualitative benefits of SAMGs. The SAMGs provide substantial defense-in-depth benefits associated with their use following the onset of core damage. These benefits are discussed in detail in appendix A to the draft regulatory analysis. Based on the NRC's assessment of the costs and benefits of the proposed rule, the NRC has concluded that the proposed rule is justified. For more information, please see the draft regulatory analysis (Agencywide Document Access and Management System (ADAMS) Accession No. ML15049A212).

The NRC prepared a draft supplemental regulatory analysis to determine the expected quantitative costs and benefits of the proposed requirement for new reactors to include design features in the plant design sufficient to enhance coping duration and minimize reliance on human actions to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities during an extended loss of all ac power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. The supplemental regulatory analysis concludes that the proposed requirements are justified and consistent with the Commission's expectations for new reactors. For more information, please see the draft supplemental regulatory analysis (ADAMS Accession No. ML15069A278).

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I. Obtaining Information and Submitting Comments.

A. Obtaining Information.

Please refer to Docket ID NRC-2014-0240 when contacting the U.S. Nuclear Regulatory Commission (NRC) about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2014-0240.
- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public Documents" and then select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. For the convenience of the reader, instructions about obtaining materials referenced in this document are provided in the "Availability of Documents" section.
- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

B. Submitting Comments.

Please include Docket ID NRC-2014-0240 in the subject line of your comment submission.

The NRC cautions you not to include identifying or contact information in comment submissions that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at http://www.regulations.gov as well as enter the comment submissions into ADAMS, and the NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment into ADAMS.

You may submit comments on the information collections by the methods indicated in the Paperwork Reduction Act Statement.

II. Background.

A. Fukushima Dai-ichi.

At 2:46 p.m. Japan standard time on March 11, 2011, the Great East Japan Earthquake, rated a magnitude 9.0, occurred at a depth of approximately 25 kilometers, 130 kilometers east of Sendai and 372 kilometers northeast of Tokyo off the coast of Honshu Island. This earthquake resulted in the automatic shutdown of 11 nuclear power plants (NPPs) at four sites along the northeast coast of Japan including the three reactors at the Fukushima Dai-ichi NPP (the three remaining plants were in outages). The earthquake precipitated a large tsunami that is estimated to have exceeded 14 meters in height at the Fukushima Dai-ichi NPP. The earthquake and tsunami produced widespread devastation across northeastern Japan, resulting in approximately 25,000 people dead or missing, displacing many tens of thousands of people, and significantly impacting the infrastructure and industry in the northeastern coastal areas of Japan.

The earthquake and tsunami disabled the majority of the external and internal electrical power systems at the Fukushima Dai-ichi NPP, leaving it with only a few hours' worth of battery power. Since a NPP licensee typically relies on electrical power to keep its reactor core and spent fuel pool (SFP) cool, this loss of internal and external power was a significant challenge to operators at Fukushima Dai-ichi. In addition, the combination of severe events challenged the implementation of emergency plans and procedures.

B. NRC Near-Term Task Force.

The NRC Chairman's tasking memorandum, COMGBJ-11-0002, "NRC Actions Following the Events in Japan," established a senior-level task force referred to as the "Near-Term Task Force" (NTTF) to conduct a systematic and methodical review of NRC regulations and processes to determine if the agency should make safety improvements in light of the events in Japan. On July 12, 2011, the NRC staff provided the Commission with the report of the NTTF (NTTF Report) as an enclosure to SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan." The NTTF concluded that continued U.S. plant operation and NRC licensing activities present no imminent risk to public health and safety. While the NTTF also concluded that the current regulatory system has served the NRC and the public well, it found that enhancements to safety and emergency preparedness are warranted and made a dozen general recommendations for Commission consideration. In examining the Fukushima Dai-ichi accident for insights for reactors in the United States, the NTTF addressed protecting against accidents resulting from natural phenomena, mitigating the consequences of such accidents, and ensuring emergency preparedness. The NTTF found that the Commission's longstanding defense-in-depth philosophy, supported and modified as necessary by state-of-the-art probabilistic risk assessment techniques, should continue to serve as the primary organizing principle of its regulatory framework. The NTTF concluded that the application of the defense-in-depth philosophy could be strengthened by including explicit requirements for beyond-design-basis events.

In response to the NTTF Report, the Commission directed the NRC staff to engage with stakeholders to review and assess the NTTF recommendations in a comprehensive and holistic manner and to provide the Commission with fully-informed options and recommendations. The Commission's Staff Requirements Memorandum (SRM)-SECY-11-0093 provided that direction and specifically directed the NRC staff to pursue recommendation 1 of the NTTF Report independent of the activities associated with the review of the remaining recommendations. The NTTF's recommendation 1 was to establish a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations. This recommendation included steps for the establishment of a Commission policy statement for a risk-informed defense-in-depth framework including extended design-basis requirements and the initiation of rulemaking to implement that framework. The results of the NRC staff work on NTTF recommendation 1 were provided to the Commission in SECY-13-0132, "Plan for Updating the U.S. Nuclear Regulatory Commission's Cost Benefit Guidance," and dispositioned by the Commission in SRM-SECY-13-0132, which specifically disapproved the establishment of a design-basis extension category of events and associated regulatory requirements, but allowed for reevaluation, as appropriate, in the context of the

Commission direction on the proposed policy statement for a long-term Risk Management Regulatory Framework. That work is outside of the scope of this rulemaking.

C. Implementation of the NTTF Recommendations.

Following the issuance of the NTTF Report, the NRC staff provided the Commission with recommendations for near-term action in SECY-11-0124, “Recommended Actions to be Taken Without Delay from the Near-Term Task Force Report,” dated September 9, 2011. The suggested near-term actions addressed several NTTF recommendations associated with this rulemaking including NTTF recommendations 4, 8, and 9.3. In SRM-SECY-11-0124, dated October 18, 2011, the Commission directed the NRC staff to, among other things: initiate a rulemaking to address NTTF recommendation 4, Station Blackout (SBO) regulatory actions, as an Advance Notice of Proposed Rulemaking (ANPR); designate the SBO rulemaking associated with NTTF recommendation 4 as a high priority rulemaking; craft recommendations that continue to realize the strengths of a performance-based system as a guiding principle; and consider approaches that are flexible and able to accommodate a diverse range of circumstances and conditions. As discussed more fully in later portions of this proposed rule, the regulatory actions associated with NTTF recommendation 4 evolved substantially from this early Commission direction, and included issuance of Order EA-12-049 that, as implemented, ultimately addressed all of NTTF recommendation 4 as well as other recommendations.

In SECY-11-0137, “Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned,” dated October 3, 2011, the NRC staff, based on its assessment of the NTTF recommendations, proposed to the Commission a three-tiered prioritization for implementing regulatory actions stemming from the NTTF recommendations. The Tier 1 recommendations were those actions having the greatest safety benefit that could be implemented without unnecessary delay. The Tier 2 recommendations were those actions that needed further technical assessment or critical skill sets to implement, and the Tier 3 recommendations were longer-term actions that depended on the completion of a shorter-term action or needed additional study to support a regulatory action. On December 15, 2011, the Commission approved the staff's recommended prioritization in SRM-SECY-11-0137.

The NTTF recommendations that form the basis of this rulemaking activity are:

- NTTF recommendation 4: strengthen SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events;
- NTTF recommendation 7: enhance spent fuel pool makeup capability and instrumentation for the spent fuel pool;
- NTTF recommendation 8: strengthen and integrate onsite emergency response capabilities such as emergency operating procedures (EOPs), Severe Accident Management Guidelines (SAMGs), and extensive damage mitigation guidelines (EDMGs);
- NTTF recommendation 9: require that facility emergency plans address staffing, dose assessment capability, communications, training and exercises, and equipment and facilities for prolonged station blackout, multi-unit events, or both;
- NTTF recommendation 10: pursue additional emergency protection topics related to multi-unit events and prolonged station blackout, including command and control structure and the qualifications of decision makers; and
- NTTF recommendation 11: pursue emergency management topics related to decision making, radiation monitoring, and public education, including the ability to deliver equipment to the site with degraded offsite infrastructure.

In response to input received from stakeholders, the NRC accelerated the schedule originally proposed in SECY-11-0137. On February 17, 2012, the NRC staff recommended in

SECY-12-0025, “Proposed Orders and Requests for Information in Response to Lessons Learned From Japan’s March 11, 2011, Great Tōhoku Earthquake and Tsunami,” that the Commission issue orders and requests for information.

To address Tier 1 NTTF recommendation 4, the NRC issued Order EA-12-049 on March 12, 2012, requiring all U.S. nuclear power plant licensees to implement strategies that would allow them to cope without their permanent electrical power sources for an indefinite period of time. These strategies would provide additional capability to maintain or restore reactor core and spent fuel cooling, as well as protect the reactor containment. This order also addressed: portions of NTTF recommendation 9 to require that facility emergency plans address prolonged station blackouts and multi-unit events; portions of NTTF recommendation 10 to pursue additional emergency protection topics related to multi-unit events and prolonged station blackout; and portions of NTTF recommendation 11 to pursue emergency procedure topics related to decision-making, radiation monitoring, and public education.

To address Tier 1 NTTF recommendation 7, the NRC issued Order EA-12-051 on March 12, 2012, requiring all U.S. nuclear power plant licensees to have a reliable indication of the water level in associated spent fuel storage pools.

To address Tier 1 NTTF recommendation 8, the NRC issued an ANPR on April 18, 2012 (77 FR 23161), to engage stakeholders in rulemaking activities associated with the methodology for integration of onsite emergency response processes, procedures, training and exercises.

D. Consolidation of Regulatory Efforts.

While developing the NTTF rulemakings, the NRC staff recognized that efficiencies could be gained by consolidating the rulemaking efforts due to the inter-relationships among the proposed changes. The NRC staff recommended to the Commission in COMSECY-13-0002, "Consolidation of Japan Lessons Learned Near-Term Task Force Recommendations 4 and 7

Regulatory Activities," COMSECY-13-0010, “Schedule and Plans for Tier 2 Order on Emergency Preparedness for Japan Lessons Learned,” and SECY-14-0046, “Fifth 6-Month Status Update on Response to Lessons Learned From Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami,” the consolidation of rulemaking activities that address NTTF recommendations 4, 7, 8, portions of 9, 10.2, and 11.1. Section II.B of this document contains a more complete discussion of the scope of NTTF recommendations addressed by this proposed rule. The Commission approved these consolidations in the associated SRMs. These consolidations were intended to:

1. Align the proposed regulatory framework with ongoing industry implementation efforts to produce a more coherent and understandable regulatory framework. Given the complexity of these requirements and their associated implementation, the NRC concluded that this is an important objective for the regulatory framework.
2. Reduce the potential for inconsistencies and complexities between the related rulemaking actions that could occur if the efforts remained as separate rulemakings.
3. Facilitate better understanding of the proposed requirements for both internal and external stakeholders, and thereby lessen the impact on internal and external stakeholders who would otherwise need to review and comment on multiple rulemakings while cross-referencing both proposed rules and sets of guidance documents.

E. Public Involvement.

This proposed rule consolidates two previous rulemaking efforts: the Station Blackout Mitigation Strategies rulemaking, directed by SRM-COMSECY-13-0002, and the Onsite Emergency Response Capabilities rulemaking, which implemented NTTF recommendation 8. Both regulatory efforts offered extensive external stakeholder involvement opportunities, including public meetings, ANPRs issued for public comment, and draft regulatory basis documents issued for public comment. The major opportunities for stakeholder involvement were:

1. Station Blackout ANPR (77 FR 16175; March 20, 2012);
2. Onsite Emergency Response Capabilities ANPR (77 FR 23161; April 18, 2012);
3. Station Blackout Mitigation Strategies draft regulatory basis and draft rule concepts (78 FR 21275; April 10, 2013). The final Station Blackout Mitigation Strategies regulatory basis was subsequently issued on July 23, 2013 (78 FR 44035); and
4. Onsite Emergency Response Capabilities draft regulatory basis (78 FR 1154; January 8, 2013). The final Onsite Emergency Response Capabilities regulatory basis, with preliminary proposed rule language, was subsequently issued on October 25, 2013 (78 FR 63901).

The NRC described in each final regulatory basis document how it considered stakeholder feedback in developing the respective final regulatory basis, including consideration of ANPR comments and draft regulatory basis document comments. Section 5 of the Station Blackout Mitigation Strategies regulatory basis document includes a discussion of stakeholder feedback used to develop the final regulatory basis. Appendix B to the Onsite Emergency Response Capabilities regulatory basis includes a discussion of stakeholder feedback used to develop that final regulatory basis.

The public has had multiple opportunities to engage in these regulatory efforts. Most noteworthy were the following:

1. Preliminary proposed rule language for Onsite Emergency Response Capabilities made available to the public on November 15, 2013 (78 FR 68774).
2. Consolidated rulemaking proof of concept language made available to the public on February 21, 2014.

3. Preliminary proposed rule language for Mitigation of Beyond-Design-Basis Events rulemaking made available to the public on August 15, 2014.
4. Preliminary proposed rule language for Mitigation of Beyond-Design-Basis Events rulemaking made available to the public on November 13, 2014 and December 8, 2014 to support public discussion with the Advisory Committee on Reactor Safeguards (ACRS).

The NRC staff has had numerous interactions with the ACRS, and in all cases these were public meetings, including the following:

1. The ACRS Plant Operations and Fire Protection subcommittee met on February 6, 2013 to discuss the Onsite Emergency Response Capabilities regulatory basis.
2. The ACRS Regulatory Policies and Practices subcommittee met on December 5, 2013 and April 23, 2013 to discuss the Station Blackout Mitigation Strategies regulatory basis.
3. The ACRS full committee met on June 5, 2013 to discuss the Station Blackout Mitigation Strategies regulatory basis.
4. The ACRS Fukushima subcommittee met on June 23, 2014 to discuss consolidation of Station Blackout Mitigation Strategies and Onsite Emergency Response Capabilities rulemakings.
5. The ACRS full committee met on July 10, 2014 to discuss consolidation of Station Blackout Mitigation Strategies and Onsite Emergency Response Capabilities rulemakings.
6. The ACRS Fukushima subcommittee met on November 21, 2014 to discuss preliminary proposed Mitigation of Beyond-Design-Basis Events rulemaking language.
7. The ACRS Fukushima full committee met on December 4, 2014 to discuss preliminary proposed Mitigation of Beyond-Design-Basis Events rulemaking language.

The NRC held many additional public meetings that have supported the development of this proposed rule. Not withstanding these efforts to engage the public during the preparation of this proposed rule, the Commission is committed to the rigors of the notice-and-comment process enacted by the Administrative Procedures Act, and is providing members of the public a 75-day comment period on the requirements NRC is proposing today.

III. Petitions for Rulemaking.

During development of this proposed rule, the NRC gave consideration to the issues raised in six petitions for rulemaking (PRMs) submitted to the NRC, five from the National Resources Defense Council Inc. (NRDC) (PRM-50-97, PRM-50-98, PRM-50-100, PRM-50-101, and PRM-50-102), and one submitted by Mr. Thomas Popik (PRM-50-96). The petitions filed by NRDC use the NTTF Report as the sole basis for the PRMs. The NTTF recommendations that the NRDC PRMs rely upon are: 4.1, 7.5, 8.4, 9.1, and 9.2. This proposed rule addresses each of these recommendations, and therefore it would resolve the issues raised by the PRMs. The NRDC petitions were dated July 26, 2011, and docketed by the NRC on July 28, 2011. The NRC published a notice of receipt in the *Federal Register* on September 20, 2011 (76 FR 58165), and did not ask for public comment at that time.

In PRM-50-97 (NRC-2011-0189), the NRDC requested emergency preparedness enhancements for prolonged station blackouts in the areas of communications ability, Emergency Response Data System (ERDS) capability, training and exercises and equipment and facilities (NTTF recommendation 9.2). The NRC determined that the issues raised in this PRM should be considered in the NRC's rulemaking process. The NRC's consideration of the issues raised in PRM-50-97 are reflected in the proposed provisions in § 50.155(e) and (f), and the proposed amendments to appendix E in both section VI and in new section VII, "Communications and Staffing Requirements for the Mitigation of Beyond Design Basis Events."

The NRC concludes that consideration of the PRM issues, as discussed herein, will address PRM-50-97. The NRC is closing the docket for this petition and intends to take final action on this petition in the Federal Register notice the NRC issues for the final Mitigation of Beyond-Design-Basis Events rule.

In PRM-50-98 (NRC-2011-0189), the NRDC requested emergency preparedness enhancements for multi-unit events in the areas of personnel staffing, dose assessment capability, training and exercises, and equipment and facilities (NTTF recommendation 9.1). The NRC determined that the issues raised in this PRM should be considered in the NRC's rulemaking process. The NRC's consideration of the issues raised in PRM-50-98 are reflected in the proposed provisions § 50.155(b)(5), (e), and (f); and the proposed amendment to appendix E in section IV as well as the addition of a new section VII. The NRC concludes that consideration of the PRM issues, as discussed herein, will address PRM-50-98. The NRC is closing the docket for this petition and intends to take final action on this petition in the Federal Register notice the NRC issues for the final Mitigation of Beyond-Design-Basis Events rule.

In PRM-50-100, the NRDC requested enhancement of spent fuel pool makeup capability and instrumentation for the spent fuel pool (NTTF recommendation 7.5). The NRC determined that the issues raised in this PRM should be considered in the NRC's rulemaking process, and the NRC published a document in the *Federal Register* with this determination on July 23, 2013 (78 FR 44034). The NRC's consideration of the issues raised in PRM-50-100 are reflected in the proposed provisions § 50.155(b)(1) and (c)(4). The proposed rule would make generically applicable NRC's Order EA-12-051, "Spent Fuel Pool Instrumentation." The NRC concludes that consideration of the PRM issues, as discussed herein, addresses PRM-50-100. The NRC has already closed the docket for this petition and intends to take final action on this petition in the Federal Register notice the NRC issues for the final Mitigation of Beyond-Design-Basis Events rule.

In PRM-50-101, the NRDC requested that § 50.63, “Loss of all alternating current power,” (sic) be revised to establish a minimum coping time of 8 hours for a loss of all alternating current (ac) power, establish the equipment, procedures, and training necessary to implement an extended loss of ac power (72 hours) for core and spent fuel pool cooling and for reactor coolant system and primary containment integrity as needed, and preplan/prestage offsite resources to support uninterrupted core and spent fuel pool cooling and reactor coolant system and containment integrity as needed (NTTF recommendation 4.1). The NRC determined that the issues raised in this PRM should be considered in the NRC’s rulemaking process, and the NRC published a document in the *Federal Register* with this determination on March 21, 2012, (77 FR 16483). The NRC’s consideration of the issues raised in PRM-50-101 is reflected in the proposed provisions § 50.155(b)(1), (c), (e), (f), and (g). The NRC concludes that consideration of the PRM issues, as discussed herein, addresses PRM-50-101. The NRC has already closed the docket for this petition and intends to take final action on this petition in the Federal Register notice the NRC issues for the final Mitigation of Beyond-Design-Basis Events rule.

In PRM-50-102, the NRDC requested more realistic, hands-on training and exercises on SAMGs and EDMGs for licensee staff expected to implement those guideline sets and make decisions during emergencies (NTTF recommendation 8.4). The NRC determined that the issues raised in this PRM should be considered in the NRC’s rulemaking process, and the NRC published a document in the *Federal Register* with this determination on April 27, 2012 (77 FR 25104). The NRC’s consideration of the issues raised in PRM-50-102 are reflected in the proposed provisions § 50.155(e) and (f). The NRC concludes that consideration of the PRM issues, as discussed herein, addresses PRM-50-102. The NRC has already closed the docket for this petition and intends to take final action on this petition in the Federal Register notice the NRC issues for the final Mitigation of Beyond-Design-Basis Events rule.

In PRM-50-96, Mr. Thomas Popik requested that the NRC amend its regulations to require facilities licensed by the NRC to assure long-term cooling and unattended water makeup of spent fuel pools in the event of geomagnetic storms caused by solar storms resulting in long-term losses of power. The NRC determined that the issues raised in this PRM should be considered in the NRC's rulemaking process and the NRC published a document in the *Federal Register* with this determination on December 18, 2012 (77 FR 74788). In that *Federal Register* document, the NRC also closed the docket for this petition. Specifically, the NRC indicated that it would monitor the progress of the mitigation strategies rulemaking to determine whether the requirements established would address, in whole or in part, the issues raised in the PRM. In this context, the proposed requirements in § 50.155(b)(1) and (c) and the associated draft regulatory guidance should address in part the issues raised because these actions would establish offsite assistance to support maintenance of the key functions (including both reactor and spent fuel pool cooling) following an extended loss of ac power that has been postulated for geomagnetic events. Additional consideration of these issues will result from NRC's participation in the interagency task force developing a National Space Weather Strategy and the associated action plan. Both the strategy and action plan are expected to be completed in 2015. When the National plans are completed, the NRC will reevaluate the need for additional actions to address the impact of geomagnetic storms on nuclear power plants within the overall context of the National Space Weather Strategy and action plan.

IV. Discussion.

A. Rulemaking Objectives.

The regulatory objectives of this rulemaking are to: 1) make the requirements in Order EA-12-049 and Order EA-12-051 generically applicable, giving consideration to lessons learned from implementation of the orders; 2) establish new requirements for an integrated response capability that includes SAMGs; 3) establish new requirements for actions that are related to onsite emergency response; 4) provide requirements for mitigating strategies for new reactor designs; and 5) address issues raised by PRMs that were submitted in the aftermath of the March 2011 Fukushima Dai-ichi event.

1. Make the requirements in Order EA-12-049 and Order EA-12-051 generically applicable, giving consideration to lessons learned from implementation of the orders.

An objective of this rulemaking is to place the requirements in Order EA-12-049 and Order EA-12-051 into the NRC's regulations so that they apply to all current and future power reactor applicants, and to provide regulatory clarity and stability to power reactor licensees. In making the requirements of Order EA-12-049 generically-applicable, this proposed rule would also consider the reevaluated hazard information developed in response to the March 12, 2012, NRC letter issued under § 50.54(f) as part of providing reasonable protection for mitigation strategies equipment against external flooding or seismic hazards. Because these orders were issued to current licensees, the requirements of these orders would not apply to future licensees. In the absence of this proposed rule, these requirements would need to be implemented for new reactor applicants or licensees through additional orders or license conditions (as was done for the Vogtle Electric Generating Plant, Units 3 and 4 and Virgil C. Summer Nuclear Station, Units 2 and 3 combined licenses (COLs), respectively). As part of the rulemaking, the NRC considered stakeholder feedback and lessons-learned from the implementation of the orders, including any challenges or unintended consequences associated with implementation. The NRC reflected this stakeholder input in the draft regulatory guidance for this proposed rule.

2. Establish new requirements for an integrated response capability that includes SAMGs.

An objective of this rulemaking is to establish requirements for an integrated response capability for beyond-design-basis events that would integrate existing strategies and guidelines (implemented through guideline sets) with the existing EOPs. This would include guideline sets that implement the requirements of current § 50.54(hh)(2) and Order EA-12-049. This would also include guideline sets that implement SAMGs, which are currently a voluntary industry initiative. This proposed rule would require sufficient staffing, command and control, training, drills, and change control to support the integrated response capability.

3. Establish new requirements for actions that are related to onsite emergency response.

An objective of this rulemaking is to establish requirements for onsite emergency response capabilities being implemented in conjunction with the implementation of Order EA-12-049. This proposed rule contains new requirements for staffing and communications assessment, and clarifies requirements for multi-source dose assessment.

4. Provide requirements for mitigating strategies for new reactor designs.

An objective of this rulemaking is to establish requirements for applicants for new reactor designs to provide mitigating strategies for beyond-design-basis events. This proposed rule contains requirements for including design features in the plant design sufficient to enhance coping durations and minimize reliance on human actions to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities during an extended loss of all ac power (ELAP) concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink.

5. Address a number of PRMs submitted in the aftermath of the March 2011 Fukushima Dai-ichi event.

An objective of this rulemaking is to address the five PRMs filed by the NRDC that raise issues that pertain to the technical objectives of this rulemaking. The petitions rely solely on the NTTF Report, and request that the NRC undertake rulemaking in a number of areas that are addressed by this proposed rule. This proposed rule also addresses, in part, the PRM submitted by Mr. Thomas Popik.

B. Rulemaking Scope.

The scope of this rulemaking, described in terms of the relationship to various NTTF recommendations that provided the regulatory impetus for this proposed rule, includes:

1. All the requirements that were within the scope of Station Blackout Mitigation Strategies rulemaking. These requirements address NTTF recommendations 4 and 7. This aspect of this proposed rule will also address NTTF 11.1 regarding onsite emergency resources to support multi-unit events with station blackout, including the need to deliver equipment to the site despite degraded offsite infrastructure. This provision currently is being implemented through Order EA-12-049.

2. All the requirements that were within the scope of the Onsite Emergency Response Capabilities rulemaking. These requirements address NTTF recommendation 8, as directed by SRM-SECY-11-0137. This aspect of this proposed rule also addresses command and control issues in NTTF recommendation 10.2.

3. Numerous requirements regarding onsite emergency response actions are included in this proposed rule. These emergency response actions currently are being implemented by Order EA-12-049; in addition, NRC staff has developed draft guidance to

support the emergency response aspect of this proposed rule. The specific regulatory actions related to emergency response in this proposed rule and the associated NTTF recommendations are:

a. Staffing and communications: addresses NTTF recommendation 9.3; also discussed in NTTF recommendations 9.1 and 9.2. These regulatory issues currently are being implemented through Order EA-12-049. The proposed requirements also address supporting facilities and equipment, as discussed in the same NTTF recommendations.

b. Multi-source term dose assessment: addresses NTTF recommendation 9.3; also discussed in NTTF recommendation 9.1. This regulatory issue is being implemented voluntarily by industry.

c. Training and exercise: addresses NTTF recommendation 9.3; also discussed in NTTF recommendations 9.1 and 9.2. These regulatory issues currently are being implemented through Order EA-12-049.

Accordingly, this rulemaking will address all the recommendations in NTTF recommendations 4, 7, 8, 9.1, 9.2, 9.3 (with one exception - ERDS modernization is addressed, but maintenance of ERDS capability throughout the accident is not addressed), 10.2, and 11.1.

This rulemaking also addresses NTTF recommendation, 9.4: modernize ERDS. This action differs from the other regulatory actions because ERDS is not an essential component of a licensee's capability to mitigate a beyond-design-basis external event. However, ERDS is an important form of communication between the licensee and the NRC. Modernization of ERDS has been completed voluntarily by industry; therefore, NRC has included amendments to the technology-specific references in 10 CFR part 50, appendix E, section VI, "Emergency Response Data System," in this proposed rule.

Finally, this proposed rules includes a requirement for applicants for new nuclear power reactors to include design features in the plant design sufficient to enhance coping durations and minimize reliance on human actions to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities during an ELAP concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink.

Scope of Proposed SAMG Requirements

Unlike the requirements for the mitigation of beyond-design-basis external events imposed by Order EA-12-049, and requirements that address the loss of large areas of the plant due to explosions and fire in current § 50.54(hh)(2) (NRC is proposing in this rule to move these requirements to a new section), SAMGs are not an NRC requirement imposed on licensees. Nevertheless, SAMGs are well established guidance documents that have been developed by the nuclear power industry with substantial NRC involvement, have been implemented by every operating nuclear power reactor licensee for decades, and are the subject of a license condition for combined licenses. Following the Three Mile Island (TMI) accident, the nuclear power industry revised its emergency response procedures to be symptom-based, and as a result, developed EOPs. In the mid-to-late 1980s, the NRC and the nuclear power industry identified a need to consider plant conditions that could lead to a severe accident. These efforts led to the nuclear industry voluntarily initiating a coordinated program on accident management in 1990. Section 5 of Nuclear Energy Institute (NEI) 91-04 (formerly NUMARC 91-04), Revision 1, "Severe Accident Closure Guidelines," describes the elements of the industry's severe accident management closure actions. The program involves the development of: 1) a structured method by which utilities could systematically evaluate and enhance their ability to deal with potential severe accidents, 2) vendor-specific SAMGs for use by licensees in developing plant-specific SAMGs, and 3) guidance and material to support utility activities related to training for severe accidents. In 1992, the Electric Power Research Institute (EPRI) developed the SAMG

Technical Basis Report (TBR). Volume one of this report covers general actions that could be taken to manage a severe accident (referred to as SAMG candidate high level actions) and their effects, and volume two is a detailed report on the physics of accident progression. By letter dated June 20, 1994, the NRC accepted the industry's approach for mitigating the consequences of severe accidents, including licensee regulatory commitments to implement plant-specific SAMGs, using the guidance developed in section 5 of NEI 91-04, Revision 1, by December 31, 1998.

The NRC assessed the ongoing implementation of SAMGs at a select number of plants during the 1997-1998 time frame as discussed in SECY-97-132, "Status of the Integration Plan for Closure of Severe Accident Issues and the Status of Severe Accident Research," and SECY-98-131, "Status of the Integration Plan for Closure of Severe Accident Issues and the Status of Severe Accident Research," and concluded that the results of the voluntary initiative achieved the NRC's overall objectives established for accident management in SECY-89-012, "Staff Plans for Accident Management Regulatory and Research Programs." In 2012, EPRI revised the TBR to account for the initial lessons learned from the Fukushima Dai-ichi accidents, as well as enhanced understanding of severe accident behavior gained from additional research and analyses performed since the original report was published.

Following the events at Fukushima Dai-ichi, the NRC again inspected the implementation, ongoing training, and maintenance of licensee SAMGs at all power reactor sites, except those that had permanently ceased operation, through performance of Temporary Instruction (TI)-2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)." The NRC found that some licensees had not maintained the SAMGs in accordance with the latest revisions of the applicable industry generic technical guidelines nor conducted training in a consistent and systematic manner. The NRC inspectors attributed the inconsistent implementation and training on SAMGs to the voluntary nature of this initiative.

Based in part on the findings of the inspections described above, the NTTF recommended that the NRC require licensees to integrate onsite emergency response capabilities, including SAMGs. The NRC realized that before it could require licensees to integrate SAMGs with EOPs and EDMGs, it first had to require licensees to have SAMGs, as opposed to relying on the current voluntary industry initiative. Unlike the Mitigating Strategies Order requirements, which were justified as necessary for adequate protection under § 50.109, SAMGs do not involve adequate protection. Because the imposition of SAMGs also would not be necessary to bring licensees into compliance with an existing NRC requirement, a SAMGs requirement would have to be justified under § 50.109 as a cost-justified, substantial increase in protection of the public health and safety or common defense and security.

As part of the effort to develop the backfitting justification for imposition of SAMG requirements, the NRC sought to make use of any applicable quantified risk information that might provide risk insights to inform the justification. In this regard, the NRC looked at its recent technical analysis[1] performed in support of the Containment Protection and Release Reduction (CPRR) rulemaking regulatory basis[2]. This analysis is relevant because it examined regulatory alternatives that would be implemented after core damage to determine whether any of the contemplated approaches can be justified under the NRC's backfitting provisions. In this respect, the risk insights stemming from this work might have relevance to NRC's consideration of SAMG requirements where the safety benefits would occur after core damage. The NRC also considered other post-Fukushima regulatory efforts (e.g., the safety benefits due to

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[1] The technical risk insights were presented to the ACRS Reliability and PRA, and Fukushima subcommittees on August 22, 2014, and to the ACRS Reliability and PRA subcommittee on November 19, 2014. This footnote is informational only; it does not imply advisory committee endorsement of the technical analysis.
[2] Refer to the draft regulatory basis for Containment Protection and Release Reduction.

implementation of Order EA-12-049 mitigation strategies, which result in a reduction in core damage frequency) within this technical analysis. The NRC acknowledges that the work to support the CPRR rulemaking was not conducted to provide a complete quantitative measure of the possible safety benefits of SAMG requirements, particularly with regard to how SAMGs might benefit maintenance of containment integrity or support more informed protective action recommendations by the emergency response organization following core damage. However, this technical analysis work does provide valuable risk insights that the NRC concluded were important to fully inform the decision on this matter, and that additionally influenced the NRC's development of the proposed SAMG framework.

The CPRR technical analysis includes a screening for a conservative high estimate of frequency-weighted individual latent cancer fatality risk. This screening analysis combined the highest ELAP frequency among all boiling water reactors (BWRs) with Mark I or Mark II containments, a success probability in the FLEX equipment[3] of 0.6 per demand following core melt, the highest conditional individual latent cancer fatality (ILCF) risk among all BWRs with Mark I or Mark II containments, and a worst case re-habitability assumption. This yields a conservative high estimate of frequency-weighted individual latent cancer fatality risk of approximately 7×10^{-8} per reactor year. This combination of assumptions does not exist at any BWR with a Mark I or Mark II containment. This conservative estimate of the risk can be viewed as the maximum possible risk that could be removed or reduced through regulatory action (i.e., the CPRR technical analysis examines a range of post-core damage regulatory actions for BWRs with Mark I or Mark II containments to identify whether any of these proposals might result in a safety benefit large enough to be justified under the Commission's backfitting requirements). This estimate is compared against the quantitative health objective, which is a

[3] Refer to NEI 12-06, Revision 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," for a description of industry-developed guidance on FLEX strategies and equipment.

quantitative measure that equates to 1/10 of 1 percent of the ILCF risk and relates to the Commission's Safety Goal Policy. This quantitative metric for the individual latent cancer fatality risk is approximately $2x10^{-6}$ per reactor year. This technical work shows that the risk is well below a level that equates to 1/10 of 1 percent of the surrounding population's latent cancer fatality risk. This result also means, that, from a quantitative standpoint, achieving risk reductions that might satisfy backfitting requirements is unlikely. More refined risk estimates from the same work (i.e., which remove the worst case assumptions and instead use assumptions specific to each power reactor), push this potential risk benefit significantly lower, by approximately two orders of magnitude. This result demonstrates the benefits of the NRC's regulations to both effectively keep the frequency of core damage very low at BWRs with Mark I and II containments, and to ensure through emergency preparedness requirements that the surrounding population is adequately protected. Those general attributes of the NRC's regulations that result in this risk insight (i.e., requirements that resulted in reduced core damage frequencies and effective emergency preparedness requirements) apply to all power reactor designs. The NRC has not performed a comprehensive quantitative analysis of the potential safety benefits of SAMG requirements for all types of reactors. However, the general risk insights obtained from the CPRR work align well with NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," (November 2012), which shows very low levels of risk (e.g., individual early fatality risk is essentially zero and ILCF risk is thousands of times lower than the NRC Safety Goal and millions of times lower than the general cancer fatility risk in the United States from all causes). As such, the available risk insights point to the likely outcome that a comprehensive quantitative analysis, where the proposed regulatory action is intended to provide its safety benefit in the post-core damage environment (as is the case for use of SAMGs), would not demonstrate a substantial safety benefit. In addition, for the specific case of the SAMG requirements in this proposed rule, the proposed regulatory action's benefit must also recognize that imposing SAMG requirements must be compared with the current regulatory state, e.g., SAMGs exist and are voluntarily in use under an industry initiative.

As discussed in the backfitting justification found in appendix A of this proposed rule's regulatory analysis, SAMG requirements would be warranted based on the small quantifiable safety benefit considered together with the much larger safety benefit from a defense-in-depth perspective, which specifically recognizes the unquantified benefits of SAMGs in support of containment integrity, particularly for beyond-design-basis external events for which uncertainties are larger and plant damage can be more severe and more readily result in core damage. The NRC developed the proposed SAMG regulatory framework in a manner that is informed by these risk insights. This proposed rule would establish requirements that would:

1. Be limited to requiring the SAMG guideline sets, and would not extend to NRC review and approval of SAMG strategies, licensee use of the equipment within the SAMGs, or licensee re-assessment of the work that industry has completed over 20 plus years to develop the SAMGs, including the recent effort to update and revise the SAMGs to reflect the Fukushima lessons learned.

2. Address the problem identified after Fukushima with the voluntary SAMG initiative, by requiring SAMGs be maintained. Specifically, this would mean that the SAMGs would be maintained within the plant configuration management system and reflect generic industry updates and improvements.

3. Result in the integration of the SAMGs with other guideline sets and with the symptom-based EOPs.

The NRC concludes this proposed regulatory structure will result in updated SAMGs that 1) reflect facility configuration, 2) include generic industry guidance, and 3) would be available for use if a severe accident occurs. As such, the SAMGs would serve as a defense-in-depth guideline set. As discussed in the "Regulatory Oversight of Severe Accident Management

Guidelines” section of this document, the NRC’s role regarding the regulatory oversight of SAMGs would be to ensure this proposed regulatory structure is in place through inspection.

Scope of Procedure and Guideline Integration

This rulemaking limits the scope of the integrated response capability to three guideline sets. This proposed rule includes these new provisions:

1. § 50.155(b)(1), resulting from Order EA-12-049, and addressing beyond-design-basis external events; these requirements are those that the NRC termed in previous regulatory basis interactions as “Station Blackout Mitigation Strategies.” The nuclear industry refers to these as “FLEX Support Guidelines” (FSGs).

2. § 50.155(b)(2) (current § 50.54(hh)(2)). These requirements are defined in NEI 06-12, Revision 2, “B.5.b Phase 2 & 3 Submittal Guideline,” as a subset of the strategies and guidelines for addressing the loss of large areas of the plant due to explosions and fires and are termed “Extensive Damage Mitigation Guidelines.” The NRC proposes to expand the scope of the generic term “EDMGs” to include all of the strategies and guidelines used to implement § 50.54(hh)(2).

3. § 50.155(b)(3), codifying the current voluntary industry initiative discussed previously in this document, typically referred to as either “SAMGs” or “Severe Accident Guidelines” (SAGs).

The NRC is proposing this integrated response capability structure to avoid unnecessarily revisiting the existing symptom-based EOPs that were developed following the TMI accident. The NRC has determined that current regulations addressing EOPs, which include the quality assurance requirements of criterion V, “Instructions, Procedures, and Drawings,” and criterion VI, “Document Control,” in appendix B to 10 CFR part 50, and the administrative controls section of the technical specifications for each plant as well as the guidance provided in regulatory guides and technical reports (e.g., NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident,” issued May 1980; NUREG-0737, “Clarification of TMI Action Plan Requirements,” issued November 1980; and NUREG–0711, “Human Factors Engineering Program Review Model,” issued November 2012) provide sufficient regulation and control of the EOPs to provide reasonable assurance of adequate protection of public health and safety. In addition, the EOPs are the subject of a national consensus standard (American National Standards Institute/American Nuclear Society 3.2 1994, “Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants.”) In order to avoid the unnecessary regulatory burden that would result by restructuring the EOPs, proposed § 50.155(b)(4) would require that the FSGs, EDMGs and SAMGs be integrated with the EOPs, rather than moving the requirements for EOPs to § 50.155.

Guideline Sets Excluded From the Proposed Rule

During the development of this proposed rule, other guideline sets were considered for inclusion within the integrated response capability. The guideline sets considered included fire response procedures, alarm response procedures (ARPs), and abnormal operating procedures (AOPs).

Similar to the EOPs, ARPs and AOPs are subject to existing NRC regulations (e.g., 10 CFR part 50, appendix B, criteria V and VI) that adequately ensure integration with other procedure sets in use at power reactors. These procedures have been used by operating power reactor licensees in actual and simulated events for many years; any further integration effort to address potential issues would likely have already been identified and corrected by existing processes (or will be identified and corrected under the quality assurance program).

The issue of whether to include fire response procedures in the scope of proposed § 50.155(b) was initially raised as recommendation 1.g. by the ACRS in its letter to the then-Chairman Jaczko dated October 13, 2011, "Initial ACRS Review of: (1) the NRC Near-Term Task Force Report on Fukushima and (2) Staff's Recommended Actions to be Taken Without Delay." That letter expressed the ACRS view that:

> [The] efforts to integrate the onsite emergency response capabilities should be expanded to include the plant fire response procedures. These procedures provide operator guidance for coping with fires that are beyond a plant's original design basis. Some plant-specific fire response procedures instruct operators to manually de-energize major electrical buses and realign fluid systems in configurations that may not be consistent with the guidance or expectations in the EOPs. Experience from actual fire events has shown that parallel execution of fire procedures, Abnormal Operating Procedures (AOPs), and EOPs can be difficult and can introduce operational complexity. Therefore, these procedures should also be included in the comprehensive efforts to better coordinate and integrate operator responses during challenging plant conditions.

This recommendation was reiterated in the ACRS letter of November 8, 2011, "ACRS Review of Staff's Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned (SECY-11-0137)."

In SECY-12-0025, enclosure 3, the NRC documented the formal process used in evaluating additional recommendations that were made by the ACRS as follows:

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

By letter dated February 27, 2012, the NRC responded to the ACRS recommendations of October 13, 2011 and November 8, 2011, discussing the disposition of ACRS recommendation 1.g. as follows:

> The NRC staff evaluated how to appropriately integrate the fire response procedure into a licensee's onsite emergency response capabilities and determined that the fire response procedures would be best considered with the agency's Tier 3 actions associated with NTTF Recommendation 3.

This disposition of the ACRS recommendation also was documented in SECY-12-0025. In its letter of March 13, 2012, the ACRS acknowledged that the formal screening process used by the NRC for additional recommendations was acceptable, but nevertheless expressed the view that "[i]ntegration of the fire response procedures presents similar challenges to those faced by integration of the SAMGs and EDMGs with the EOPs. Hence, this integration effort should be addressed as part of NTTF Recommendation 8 rather than as a seismic-induced-fire issue under NTTF Recommendation 3."

Recognizing the continued ACRS interest in the integration of fire response procedures with onsite emergency actions and the existence of an additional program of work to be taken up on the ACRS recommendation, the NRC has concluded that the reasoning underlying the initial prioritization of ACRS recommendation 1.g was sound and it would be inappropriate to include fire response procedure integration within this rulemaking effort. The NRC offers the following reasons for the exclusion of firefighting strategies and procedures from the scope of integration in this rulemaking:

1. The NRC-required fire protection program is designed to function autonomously from other ongoing activities and is implemented by a fire brigade that is manned in all modes of operation and is well-trained. Firefighting activities are led by personnel knowledgeable of overall plant operations, including the equipment necessary for safe shutdown of the plant.

These personnel communicate with the main control room in order to prioritize and deconflict activities.

2. Comprehensive firefighting strategies and implementing procedures have been developed for each area of the plant and fire brigade qualified individuals participate in drills on a quarterly basis to demonstrate proficiency with the use of these strategies and procedures in the context of concurrent use of other, non-integrated procedures throughout the plant.

3. EOPs, EDMGs, FSGs, and SAMGs account for equipment lost due to concurrent fires during events by providing alternate methods to accomplish the functions the equipment was to have performed.

C. Proposed Rule Construct

To accomplish NRC's rulemaking objectives in a manner consistent with the described scope, this proposed rule has been based on these precepts:

1. The central requirement will be an integrated response capability that includes currently existing procedures and guideline sets, and will add SAMGs. Additional requirements will support this integrated response capability.

2. The mitigation strategies under Order EA-12-049 established the basic framework for broader capability to mitigate beyond-design-basis external events that impact an entire reactor site. This framework includes: supporting drills, training, change control, staffing, communications capability, multi-source term assessment capability, and command and control.

3. Integration of this capability into NRC's requirements has been extended to include the post-core-damage part of accident mitigation (i.e., SAMGs).

As a result, the proposed new § 50.155 is structured to have:

1. Integrated response requirements in paragraph (b).

2. Supporting equipment requirements in paragraph (c) that include equipment required by both Order EA-12-049 and Order EA-12-051.

3. External hazard equipment protection requirements in paragraph (c) that reflect the hazard information developed under the § 50.54(f) letter of March 12, 2012.

4. Design features requirements for applicants for new reactor designs in paragraph (d).

5. Supporting training, drills, and change control requirements in paragraphs (e), (f), and (g).

6. Requirements that relate to enhanced onsite emergency response capabilities located in appendix E to 10 CFR part 50, to include a new section VII.

7. Implementation requirements that establish compliance deadlines in paragraph (h).

8. Supporting new reactor application requirements (under either 10 CFR part 50 or part 52 processes) in the appropriate content of applications portions.

The proposed rule is applicable to both current licensees and new applicants (under either 10 CFR part 50 or part 52) as established by proposed paragraph (a). Additionally, the proposed rule contains provisions to facilitate power reactor decommissioning.

D. Proposed Rule Regulatory Bases.

Applicability

The proposed rule would apply, in whole or in part, to applicants for and holders of a construction permit or operating license for a nuclear power reactor under 10 CFR part 50, or combined license under 10 CFR part 52, and applicants for standard design certifications, standard design approvals, and manufacturing licenses under 10 CFR part 52.

This proposed rule would not apply to holders of an operating license for a non-power reactor under 10 CFR part 50. Non-power reactor licensees would not be subject to this proposed rule because non-power reactors pose lower radiological risks to the public from accidents than do power reactors because: 1) the core radionuclide inventories in non-power reactors are lower than in power reactors as a result of their lower power levels and often shorter operating cycle lengths; and 2) non-power reactors have lower decay heat associated with a lower risk of core melt and fission product release in a loss-of-coolant accident than power reactors.

A holder of a general or specific 10 CFR part 72 independent spent fuel storage installation (ISFSI) license would not be subject to this proposed rule for the ISFSI, because the decay heat load of the irradiated fuel would be sufficiently low prior to movement to dry cask storage that it could be air-cooled. This would meet the proposed sunsetting criteria (discussed later in this section of this document,).

The GE Morris facility in Illinois consists of a spent fuel pool licensed under 10 CFR part 72 as an ISFSI and consequently would not need to comply with this proposed rule. The NRC considered including the GE Morris facility within the scope of this rule but found that the age (and corresponding low decay heat load) of the fuel in the facility made it unnecessary. The GE Morris facility also would meet this proposed rule’s sunsetting criteria. While the proposed rule would leave in force the requirements of the current § 50.54(hh)(2), those requirements are not applicable to GE Morris due to its status as a non-part 50 licensee. In the course of the development and implementation of the guidance and strategies required by the current § 50.54(hh)(2), the NRC evaluated whether additional mitigation strategies were warranted at GE Morris and concluded that no mitigating strategies were warranted beyond existing measures, due to the extended decay time since the last criticality of the fuel stored there, the resulting low decay heat levels, and the assessment that a gravity drain of the GE

Morris SFP is not possible due to the low permeability of the surrounding rock and the high level of upper strata groundwater.

This proposed rule would establish a “sunsetting” or phased removal of requirements for licensees of decommissioning power reactors. Licensees would not need to meet requirements that relate to the reactor source term and associated fission product barriers once all fuel has been permanently removed from the reactor vessel and placed in the spent fuel pool. This proposed rule would require secondary containment for reactor designs that employ this feature as a fission product barrier for the spent fuel pool source term.

Once the NRC has docketed a licensee’s § 50.82(a)(1) or § 52.110(a) certification of permanent removal of fuel from the reactor vessel and certification of permanent cessation of operations, that licensee would not be subject to requirements to have mitigation strategies and guidelines for maintaining or restoring core cooling and containment capabilities. As discussed previously, these proposed requirements are based on Order EA-12-049. The licensees for the Kewaunee Power Station, Crystal River Unit 3 Nuclear Generating Plant, San Onofre Nuclear Generating Station Units 2 and 3, and Vermont Yankee Nuclear Power Station, submitted § 50.82(a)(1) certifications after issuance of Order EA-12-049; the NRC has recinded Order EA-12-049 to this group of NPP licensees (Shutdown NPP Group). These rescissions were based on the NRC’s conclusion that the lack of fuel in the licensee’s reactor core and the absence of challenges to the containment rendered unnecessary the development of guidance and strategies to maintain or restore core cooling and containment capabilities. Consistent with these rescissions, the NRC proposes to relieve licensees in decommissioning from the requirement to comply with proposed requirements to have mitigation strategies and guidance to maintain or restore core cooling and containment capabilities. Moreover, these licensees would not need to comply with any of the requirements in this proposed rule based on compliance with the proposed requirement to have mitigation strategies and guidelines for maintaining or restoring core cooling and containment capabilities.

This proposed rule treats the EDMG and SAMG requirements in a manner similar to the requirements for FSGs. For a licensee who has § 50.82(a)(1) or § 52.110(a) certifications docketed at the NRC, the lack of fuel in their reactor core and the absence of challenges to the containment would render unnecessary EDMGs for core cooling and containment capabilities and SAMGs related to fuel in the reactor vessel. This licensee would not need to comply with any of the requirements in this proposed rule; rather, the licensee would be required to comply with the proposed requirement to have EDMGs or SAMGs as based on the presence of fuel in the reactor vessel.

Once the NRC has docketed a licensee's § 50.82(a)(1) or § 52.110(a) certifications, that licensee would not need to comply with the requirements proposed by this rule that the equipment relied on for the mitigation strategies include reliable means to remotely monitor wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions. These proposed requirements are based on the requirements in Order EA-12-051. This order requires a reliable means of remotely monitoring wide-range SFP levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event with the potential to challenge both the reactor and SFP.

The NRC has also recinded Order EA-12-051 for the Shutdown NPP Group mentioned previously. These rescissions were based, in part, on the NRC's conclusions that once a licensee certifies the permanent removal of the fuel from its reactor vessel, the safety of the fuel in the SFP becomes the primary safety function for site personnel. In the event of a challenge to the safety of fuel stored in the SFP, decision-makers would not have to prioritize actions and the focus of the staff would be the SFP condition. Thus, once fuel is permanently removed from the reactor vessel, the basis for the Order EA-12-051 would no longer apply. Consistent with the NRC order rescissions, the NRC proposes to no longer require licensees in decommissioning to have a reliable means to remotely monitor wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event with the potential to challenge both the reactor and SFP.

Once the NRC has docketed a licensee's § 50.82(a)(1) or § 52.110(a) certifications, that licensee would not need to comply with the requirements in proposed Section VII, "Communications and Staffing Requirements for the Mitigation of Beyond Design Basis Events," in 10 CFR part 50, appendix E. These proposed requirements are based on the March 12, 2012, § 50.54(f) letters that requested operating power reactor licensees to perform, among other things, emergency preparedness communication and staffing evaluations for prolonged loss of power events consistent with NTTF recommendation 9.3. Once the licensees for the Shutdown NPP Group were no longer operating power reactors, they informed the NRC that they would no longer proceed with implementing recommendation 9.3. In response to the filings, the NRC determined that, for beyond-design-basis external events challenging the safety of the spent fuel at the Shutdown NPP Group:

> recovery and mitigation actions could be completed over a long period of time due to the slow progression of any accident as a result of the very low decay heat levels present in the pool within a few months following permanent shutdown of the reactor. Thus, spent fuel pool beyond design basis accident scenarios at decommissioning reactor sites do not require the enhanced communication and staffing that may be necessary for the reactor-centered events the 50.54(f) letter addresses.[4]

Order EA-12-049 also required power reactor licensees to have certain spent fuel pool cooling capabilities. In the rescission letters to the licensees for the Shutdown NPP Group, the

[4] See the "Availability of Documents" section of this document for the NRC letters to the licensees for Kewaunee Power Station, Crystal River Unit 3 Nuclear Generating Plant, San Onofre Nuclear Generating Station Units 2 and 3, and Vermont Yankee Nuclear Power Station.

NRC determined that, due to the passage of time, fuel's low decay heat and the long time to boil off the water inventory in the spent fuel pool obviated the need for the Shutdown NPP Group licensees to have guidance and strategies necessary for compliance with Order EA-12-049. The rescission of Order EA-12-049 for those licensees eliminated the requirement for them to comply with the Order's requirements concerning beyond-design-basis event strategies and guidelines for spent fuel pool cooling capabilities. Consistent with the basis for the Order rescissions, licensees in decommissioning could be relieved from the proposed requirements concerning beyond-design-basis event strategies and guidelines for spent fuel pool cooling capabilities and any related requirements. These licensees would have to perform and retain an analysis demonstrating that sufficient time has passed since the fuel within the spent fuel pool was last irradiated such that the fuel's low decay heat and boil-off period provide sufficient time for the licensee to obtain offsite resources to sustain the spent fuel pool cooling function indefinitely. Similarly, the proposed rule would allow removal of the SAMG requirements, leaving only the requirements for EDMGs in place. Licensees could make use of the equipment in place for EDMGs should that equipment be available, recognizing that the protection for that equipment is against the hazards posed by events that result in losses of large areas of the plant due to fires or explosions rather than beyond-design-basis external events resulting from natural phenomena. If the EDMG equipment is not available, the offsite resources would be used by the licensee for only onsite emergency response (i.e., spent fuel pool cooling). This proposed amendment would not impact any commitments licensees have made regarding exemptions from offsite emergency planning requirements, which consider a beyond-design-basis event that could result in a zirconium cladding fire due to a loss of SFP inventory and do not consider offsite resources in mitigation strategies.

The NRC proposes to maintain the EDMGs requirement, because an event for which EDMGs would be required is not based on the condition of the fuel, but may instead result from aircraft impact and a beyond-design-basis security event which could introduce kinetic energy into the spent fuel pool independent from the decay heat of the fuel. These types of events and their potential consequences were considered as a part of the rulemaking dated March 7, 2009, on Power Reactor Security Requirements (74 FR 13926). In the course of that rulemaking, the NRC took into account stakeholder input and determined that it would be inappropriate to apply the EDMG requirements to permanently shutdown and defueled reactors where the fuel was removed from the site or moved to an ISFSI. However the resulting rule was written to remove the EDMG requirements once the certifications of permanent cessation of operations and removal of fuel from the reactor vessel were submitted rather than upon removal of fuel from the SFP. The NRC proposes to correct this error from the 2009 final rule in the proposed rule as explained in the "EDMGs" portion of this section.

The NRC proposes to exclude from the proposed requirements the licensee for Millstone Power Station Unit 1, Dominion Nuclear Connecticut, Inc. Dominion Nuclear Connecticut, Inc. is also the licensee for Millstone Power Station Units 2 and 3, but this exclusion would apply to Dominion Nuclear Connecticut, Inc. in its capacity as licensee for only Unit 1, which is not operating but has irradiated fuel in its spent fuel pool and satisfies the proposed criteria for not having to comply with the proposed rule except for the EDMG requirements. In the course of the development and implementation of the guidance and strategies required by current § 50.54(hh)(2), the NRC evaluated whether additional mitigation strategies were warranted at Millstone Power Station Unit 1 and concluded that no mitigating strategies were warranted beyond existing measures, principally due to the extended decay time since the last criticality there on November 4, 1995, and the resulting low decay heat levels allowing sufficient time for the use of existing strategies augmented by mitigation strategies existing in 2005. The exclusion for Millstone Power Station Unit 1 in this proposed rule is based upon that conclusion, recognizing that additional mitigating capabilities will be present due to the implementation of the § 50.54(hh)(2) strategies at the collocated Millstone Power Station Units 2 and 3.

In contrast to Millstone Power Station Unit 1, the Shutdown NPP Group licensees were issued license conditions for the mitigating strategies corresponding to the § 50.54(hh)(2) strategies. These license conditions are condition 2.C.(10) to Renewed Operating License No. DPR-43 for Kewaunee Power Station, condition 2.C.(14) to Facility Operating License No. DPR-72 for Crystal River Unit 3 Nuclear Generating Plant, condition 2.C.(26) to Facility Operating License NPF-10 for San Onofre Nuclear Generating Station Unit 2, condition 2.C.(27) to Facility Operating License NPF-15 for San Onofre Nuclear Generating Station Unit 3, and condition 3.N to Renewed Operating License No. DPR-28 for Vermont Yankee Nuclear Power Station. Those licensees and future power reactor licensees that enter decommissioning would have the burden to show that operation in a decommissioning status with irradiated fuel in the spent fuel pool without the EDMG license condition or the proposed requirement to comply with the proposed EDMG requirement would provide adequate protection of public health and safety.

Integrated Response Capability

Each applicable applicant or licensee would be required to develop, implement, and maintain an integrated response capability that includes FSGs, EDMGs, SAMGs, EOPs, sufficient staffing, and a supporting organizational structure with defined roles, responsibilities, and authorities for directing and performing these strategies, guidelines, and procedures.

As discussed in the NTTF Report, EOPs have long been part of the NRC's safety requirements. The NRC regulations address them through the quality assurance requirements of criterion V and criterion VI in appendix B to 10 CFR part 50, and in the administrative controls section of the technical specifications for each plant. Following the accident at TMI Unit 2, EOPs were upgraded to address human factors considerations in order to improve human reliability including the operator's ability to mitigate the consequences of a broad range of initiating events and subsequent multiple failures without the need to diagnose specific events. In other words, EOPs were modified from their previous event-driven nature to be symptom-based. Numerous subsequent regulatory guides (RGs) and technical reports (e.g., NUREG-0660, NUREG-0737, and NUREG-0711) also address EOPs. In addition, the EOPs are the subject of a national consensus standard (American National Standards Institute/American Nuclear Society 3.2-2012, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"). The subject matter for the initial and requalification training, written exam, and operating test for reactor operators and senior reactor operators also includes the EOPs. While implementing EOPs, the event command and control functions remain in the control room under the direction of the senior licensed operator on shift.

The NTTF Report also discusses the development of SAMGs by the nuclear industry during the 1980s and 1990s in response to the TMI accident and follow-up activities. These follow-up activities included extensive research and study (including several probabilistic risk assessments) on severe accidents and severe accident phenomena. Depending upon the implementation of owners group guidance by particular licensees, the SAMGs were written for use by either plant technical support staff, usually located in the plant's Technical Support Center (TSC), or control room staff. The SAMGs are meant to enhance the ability of the operators to manage accident sequences that progress beyond the point where EOPs and other plant procedures are applicable and useful. The EOPs typically cover accidents to the point of imminent or actual significant fuel damage. As a result, the SAMGs extend the range of initiating events and plant damage states for which strategies and guidelines are available for use by operators beyond the procedural guidance provided by the EOPs to include circumstances that include imminent or actual significant damage to the fuel in the reactor or stored in the spent fuel pool.

The nuclear industry developed EDMGs following the terrorist events of September 11, 2001, in response to security advisories, orders, and license conditions issued by the NRC that required licensees to develop and implement guidance and strategies intended to maintain or restore core cooling and containment and spent fuel pool cooling capabilities under the circumstances associated with the loss of large areas of the plant due to fire or explosion. The EDMGs further extend the range of initiating events and plant damage states for which strategies and guidelines are available for use by operators to include the loss of large areas of the plant and a subsequent impairment of the operability and functionality of structures, systems and components that are within that area. NEI 06-12, "B.5.b Phase 2&3 Submittal Guideline," Revision 2, December 2006 (the NRC-endorsed guidance for the requirements associated with EDMGs) provides that the EDMGs "must be interfaced with existing SAMGs so that potential competing considerations associated with implementing these and other strategies are appropriately addressed."

Based upon these considerations, the NTTF recommended that the NRC require licensees to further integrate EOPs, SAMGs and EDMGs, including a clarification of transition points, command and control, decision making, and rigorous training that includes conditions that are as close to real accident conditions as feasible.

Subsequent to issuance of the NTTF Report, the range of initiating events and plant damage states for which strategies and guidelines are available for use by operators was further extended through the development of mitigating strategies for beyond-design-basis external events in response to Order EA-12-049. The development and implementation of this set of strategies and guidelines was accomplished with the knowledge of the existence of the other NTTF recommendations and took them into account to the extent practical. In order to provide better integration with the EOPs and SAMGs, the resulting strategies and guidelines (FSGs) leave the designation of command and control and decision-making functions within the EOPs or SAMGs, as appropriate. As recommended in the NTTF Report, this proposed rule would require that EDMGs, SAMGs, and FSGs be integrated with EOPs, consistent with the expectation that EOPs remain the central element of a licensee's initial response capability.

In establishing a requirement for a response capability that encompasses the use of EOPs, SAMGs, EDMGs, and FSGs, the NRC considered the fact that these strategies, guidelines and procedures were, and are currently being, developed at separate times over a period of several decades and that the associated efforts have been focused on responding to different types of initiating events and plant damage states. Although most of these strategies, guidelines and procedures were developed in response to NRC requirements, SAMGs were developed as a voluntary industry initiative. As a result, these strategies, guidelines and procedures may not properly reflect consideration of the interfaces (e.g., procedure transitions), dependencies (e.g., reliance on common systems or resources) and interactions (e.g., alignment of response strategies) among strategies, guidelines and procedures that may be used in combination, either consecutively or concurrently, to mitigate a design-basis or beyond-design-basis event.

Additionally, the NRC considered that these strategies, guidelines and procedures are not used by a single licensee organizational unit but will often require coordination and transfer of responsibilities amongst licensee organizational units. For example, although EOP actions are directed by members of the main control room (MCR) complement of licensed operators, direction for SAMG actions may begin in the MCR but transition to and be conducted principally by staff of the TSC. The EDMGs may be implemented under conditions of loss of the MCR and therefore initiated and directed by knowledgeable and available site personnel until coordination and augmentation efforts enable transition to a more stable command and control structure.

The mitigation strategies for extreme external events, though initiated by the MCR complement of licensed operators, may require coordination with and augmentation by offsite organizations. Further, and as noted previously, there are potential accident scenarios in which a licensee might employ strategies from more than one of these strategies, guidelines and procedures during its response to an accident. One plausible sequence is for an initial response to be under the EOPs, supplemented by actions under the FSGs, and ultimately transition to actions under the SAMGs. Such an accident progression would engage and require the coordination of multiple licensee organizational units.

In light of the preceding considerations, the proposed rule would require that the mitigating strategies, guidelines and procedures, staffing, and supporting organizational structure be developed, implemented, and maintained such that they function as an "integrated" response capability. The intent is to ensure that applicants and licensees establish and maintain a functional capability to produce a coordinated and logical response under a wide range of accident conditions. The intent is not to require physical integration (e.g., organizations need not be merged and strategies, guidelines and procedures need not be combined), but rather to require a functional integration of the elements of the response capability. To achieve this functional integration, the NRC expects that applicants and licensees will have addressed the interfaces, dependencies, and interactions among the elements of their response capability such that elements work together to support effective performance under the full range of accident conditions. For example, functional integration of the strategies, guidelines and procedures would ensure that transition points are explicitly identified and conflicts between strategies are eliminated to the extent practical. Functional integration of response organizations would ensure that organizations working together to use these strategies, guidelines, and procedures (e.g., to coordinate actions or provide support) have clearly defined lines of communication between the organizations, as well as clearly defined authorities and responsibilities relative to each other, such that there are no gaps or conflicts.

The proposed requirements for FSGs would make generically-applicable requirements previously imposed on licensees by Order EA-12-049 and for Virgil C. Summer Nuclear Station Units 2 and 3 by license condition as described in Memorandum and Order CLI-12-09[5]. These proposed requirements would provide additional defense-in-depth measures that increase the capability of nuclear power plant licensees to mitigate consequences of beyond-design-basis external events. Consistent with Order EA-12-049 and associated license conditions, these proposed provisions would be made generically-applicable in recognition that beyond-design-basis events have an associated significant uncertainty, and that the NRC concluded additional measures were warranted in light of this uncertainty.

The proposed FSG strategies and guideline requirements are intended to mitigate consequences of beyond-design-basis external events from natural phenomenon that result in an ELAP concurrent with either a loss of normal access to the ultimate heat sink, or for passive reactor designs, a loss of normal access to the normal heat sink. Recognizing that beyond-design-basis external events are fundamentally unbounded, and that these events can result in a multitude of damage states and associated accident conditions, a significant regulatory challenge is developing bounded requirements that meaningfully address the regulatory issue. From a practical standpoint, development of mitigation strategies requires that there be some definition (or boundary conditions established) for an onsite damage state for which the strategies would then address and thereby provide an additional capability to mitigate beyond-design-basis external event conditions that might occur. The damage state should ideally be representative of a large number of potential damage states that might occur as a result of

[5] *Summer*, CLI-12-09, 75 NRC at 440, and the V.C. Summer Unit 2 license, License No. NPF-93, Condition 2.D.(13) and V.C. Summer Unit 3 license, License No. NPF-94, Condition 2.D.(13).

extreme external events, and it should present an immediate challenge to the key safety functions, so that the resultant strategies actually improve safety. The assumed damage state for this proposed rule is the same as that assumed to implement the requirements of EA-12-049, attachment 2 for currently operating power reactors: an ELAP condition concurrent with loss of normal access to the ultimate heat sink (LUHS). This assumed damage state is effective at immediately challenging the key safety functions following a beyond-design-basis external event (i.e., core cooling, containment and spent fuel pool cooling). Requiring strategies to maintain or restore these key functions under such circumstances would result in an additional mitigation capability consistent with the Commission's objective when it issued Order EA-12-049.

The proposed rule would not be prescriptive in terms of the specific set of initial and boundary conditions assumed for the ELAP and LUHS condition, recognizing that the damage state for current operating reactors, defined in more detail in draft regulatory guidance for this proposed rule (DG)-1301, "Flexible Mitigation Strategies for Beyond-Design-Basis Events," reflects current operating power reactor designs and the reliance of those designs on ac power, while the assumed damage state for a future design may be different depending upon the design features. Specifically, this damage state was implemented through the assumption of the ELAP to the onsite emergency ac buses, but did allow for ac power from the inverters to be assumed available in order to establish event sequence and the associated times for when mitigation actions would be assumed to be required. To address the Order EA-12-049 requirement for an actual loss of all ac power, including ac power from the batteries (through inverters), contingencies are included in the mitigation strategies to enable actions to be taken under those circumstances (e.g., sending operators to immediately take manual control over a non ac-powered core cooling pump). As such, this proposed provision is meant to make generically-applicable the current implementation under EA-12-049 (i.e., there is no intent to either relax or impose new requirements), and be performance-based to allow some flexibility for future designs. As an example, some reactor designs (e.g., Westinghouse AP1000 and General Electric Economic Simplified Boiling Water Reactor (ESBWR)) use passive safety systems to meet NRC requirements for maintaining key safety functions. The inherent design of those passive safety systems makes certain assumptions, such as loss of access to the ultimate heat sink, not credible. Accordingly, the assumed condition for the FSG requirements for passive reactors is the loss of normal access to the normal heat sink, discussed further below. Nevertheless, in this proposed rule the NRC is requiring that the strategies and guidelines be capable of implementation during a loss of all ac power.

Regarding the assumed LUHS for combined licenses or applications referencing the AP1000 or the ESBWR designs, the assumption was modified to be a loss of normal access to the normal heat sink (see attachment 3 to Order EA -12-049, *Summer*, CLI-12-09, 75 NRC at 440, the V.C. Summer Unit 2 license, License No. NPF-93, Condition 2.D.(13), and the V.C. Summer Unit 3 license, License No. NPF-94, Condition 2.D.(13)). This modified language reflects the passive design features of the AP1000 and the more recently certified ESBWR that provide core cooling, containment, and spent fuel cooling capabilities for 72 hours without reliance on ac power. These features do not rely on access to any external water sources for the first 72 hours because the containment vessel and the passive containment cooling system serve as the safety-related ultimate heat sink for the AP1000 design and the isolation condenser system serves as the safety-related ultimate heat sink for the ESBWR design.

As discussed previously, the range of beyond-design-basis external events is unbounded. These proposed provisions are not intended, and should not be understood to mean that the mitigation strategies can adequately address all postulated beyond-design-basis external events. It is always possible to postulate a more severe event, with greater damages and for which the mitigation strategies may not be able to maintain or restore the functional capabilities (e.g., meteorite impact). Instead, the proposed requirements provide additional mitigation capability in light of uncertainties associated with external events, consistent with the NRC's regulatory objective when it issued Order EA-12-049.

This proposed rule would require that the FSGs be capable of being implemented site-wide. This recognizes that severe external events are likely to impact the entire reactor site, and for multi-unit sites, damage all the power reactor units on the site. This requirement means that there needs to be sufficient equipment and supporting staff to enable the core cooling, containment, and spent fuel pool cooling functions to be maintained or restored for all the power reactor units on the site. This is a distinguishing characteristic of this set of mitigating strategies from those that currently exist for § 50.54(hh)(2), for which the damage state was a more limited, albeit large area of a single plant, reflecting the hazards for which that set of strategies was developed.

The NRC gave consideration to whether there should be changes made to § 50.63 to link those requirements with this proposed rule. This consideration stemmed from recommendation 4.1 of the NTTF Report to "initiate rulemaking to revise 10 CFR 50.63" and the understanding that the proposed rule could result in an increased station blackout coping capability, in addition to the regulatory objective of the proposed provisions, which is to provide additional beyond-design-basis external event mitigation. Because of the substantive differences between the requirements of § 50.63 for licensees to be able to withstand and recover from a station blackout and the proposed requirements, the NRC determined that such a linkage was not necessary and could lead to regulatory confusion.

The principal regulatory objective of § 50.63 was to establish station blackout coping durations for a specific scenario (i.e., loss-of-offsite power coincident with a failure of both trains of emergency onsite ac power, typically, the failure of multiple emergency diesel generators.) In meeting this regulatory objective, the NRC recognized that there would be safety benefits accrued through the provision of an alternate ac source diverse from the emergency diesel generators and therefore defined such a source in § 50.2. In furtherance of this alternative means to comply with § 50.63, the NRC also defined the event a licensee must withstand and recover from as a station blackout rather than a loss of all ac power. A station blackout allows for continued availability of ac power to buses fed by station batteries through inverters or by alternate ac sources. This proposed rule would provide an additional capability to mitigate beyond-design-basis external events. Because the condition assumed for the mitigation strategies to establish the additional mitigation capability includes an ELAP, which is more conservative than a station blackout as defined in § 50.2, there can be a direct relationship between the two different sets of requirements with regard to the actual implementation at the facility. Specifically, implementation of the proposed mitigation strategies links into the station blackout procedures (e.g., the applicable strategies would be implemented to maintain or restore the key safety functions when the EOPs reach a "response not obtained" juncture).

Step-by-step procedures are not necessary for many aspects of the proposed mitigating strategies and guidelines. Rather, the strategies and guidelines should be flexible, and thus enable plant personnel to adapt them to the conditions that result from the beyond-design-basis external event. The proposed provisions typically would result in strategies and guidelines that use both installed and portable equipment, instead of only relying on installed ac power sources (with the exception of protected battery power) to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities. By using equipment that is separate from the normal installed ac-powered equipment, the strategies and guidelines have a diverse attribute. By having available multiple sets of portable equipment that can be deployed and used in multiple ways depending on the circumstances of the event, operators are able to implement strategies and guidelines that are flexible and adaptable.

The proposed mitigation strategies requirements are both performance-based and functionally-based. Performance-based requirements are proposed recognizing that the new requirements would provide most benefit to future reactors whose designs could differ significantly from current power reactor designs and as such, use of more prescriptive requirements could be problematic and create unnecessary regulatory impact and need for exemptions. Use of functionally-based requirements results from the need to have requirements that can address a wide range of damage states that might exist following beyond-design-basis external events. Maintaining or restoring three key functions (core cooling, containment and spent fuel pool cooling) supports maintenance of the fission product barriers (i.e., fuel clad, reactor coolant pressure boundary, and containment) and results in an effective means to mitigate these events, while remaining flexible such that the strategies and guidelines can be adapted to the damage state that occurs. Functionally-based requirements also result in strategies that align well with the symptom-based procedures used by power reactors to respond to accidents. Accordingly, Order EA-12-049 contained requirements for a three-phased approach for current operating reactors. This proposed rule does not specify a number of phases; instead, the NRC is proposing higher level, performance-based requirements consistent with this discussion.

The NRC gave consideration to incorporating into the proposed rule a requirement that licensees be capable of implementing the strategies and guidelines "whenever there is irradiated fuel in the reactor vessel or spent fuel pool." This provision would have been a means of making generically-applicable the requirement from Order EA-12-049 that licensees be capable of implementing the strategies and guidelines "in all modes." The NRC considers the terminology "whenever there is irradiated fuel in the reactor vessel or spent fuel pool" would be a better means to address the order requirement since it does not use technical specification type language (i.e., modes) which would not be in effect when a licensee offloads the fuel into the spent fuel pool during an outage. The NRC concluded that the use of the phrases “whenever there is irradiated fuel in the reactor vessel or spent fuel pool” or “in all modes” is not necessary because the proposed applicability provisions would ensure that licensees will be required to have mitigation strategies for beyond-design-basis external events for the various configurations that can exist for the reactor and spent fuel pools throughout the operational, refueling and decommissioning phases.

The mitigation strategies and guidelines implemented under NRC Order EA-12-049 assume a demanding condition that maximizes decay heat that would need to be removed from the reactor core and spent fuel pool source terms on site. This implementation results in a more restrictive timeline (i.e., mitigation actions required earlier following the event to take action to maintain or restore cooling to these source terms) and a greater resulting additional capability. These assumed at-power conditions are 100 days at 100 percent power prior to the event for the reactor core as was used for § 50.63. This assumption establishes a conservative decay heat for the reactor source term. The assumed spent fuel pool conditions include the design basis heat load for the spent fuel pool, typically a full core offload following a refueling outage. This establishes a conservative heat load for the spent fuel pool. The NRC recognizes that as a practical reality these conditions would not exist simultaneously. The NRC considers the development of timelines for the proposed mitigating strategies using the maximum heat load for either the reactor core or the spent fuel pool to be appropriate. While establishing the capability to mitigate the maximum heat load for both simultaneously would be compliant with the proposed requirements, it would not be necessary.

The NRC recognizes the difficulty of developing engineered strategies for the extraordinarily large number of possible plant and equipment configurations that might exist under shutdown conditions (i.e., at shutdown when equipment may be removed from service, when there is ongoing maintenance and repairs or refueling operations, or modifications are being implemented). The proposed requirements mean that licensees should be cognizant of such configurations, equipment availability, and decay heat states that could present greater challenges under these conditions, and design mitigation strategies that they can be implemented under such circumstances.

The NRC considered requiring the strategies to be developed considering the need to plan for delays in the receipt of offsite resources as a result of damage to the transportation infrastructure. While severe events could damage local infrastructure, and could create challenges with regard to the delivery of offsite resources, the NRC concluded that having this level of specificity in the proposed provisions would not be necessary. Instead this proposed rule contains provisions that are more performance-based, requiring continued maintenance or restoration of the functional capabilities until acquisition of offsite assistance and resources. Potential delays and other challenges presented by extreme events that affect acquisition and use of offsite resources would be addressed by licensee programs that implement the proposed provisions.

Order EA-12-049 included a requirement that licensees develop guidance and strategies to obtain "sufficient offsite resources to sustain [the functions of core cooling, containment, and spent fuel pool cooling] indefinitely." The NRC considered using this language in the proposed rule, but concluded that this would be better phrased as "indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies." The NRC concluded that this phrase better communicates the existence of a transition from the use of the mitigating strategies to recovery operations.

The NRC recognizes that the use of the proposed mitigating strategies would potentially require departure from a license condition or a technical specification (contained in a license issued under 10 CFR part 50 or 52) and could be considered a procedurelization of the allowance provided under § 50.54(x). Given that the initiation of the use of these strategies may be included in emergency operating procedures or other procedures, which might be considered procedures described in the final safety analysis report (as updated), there is an interaction with the provisions of § 50.59(c)(1) regarding the need to obtain a license amendment in order to make the necessary change to those procedures. The NRC considered including provisions in the proposed rule to specifically allow departures from license conditions or technical specifications in order to clarify this situation, but found these provisions unnecessary. For holders of operating licenses under 10 CFR part 50 and combined licenses under 10 CFR part 52 that were subject to Order EA-12-049, the provisions of that Order provided more specific criteria for making the necessary changes than § 50.59, making that section inapplicable as set forth in § 50.59(c)(4). Those criteria included the provision of an overall integrated plan to the NRC for review. Similar criteria were included in license conditions for the combined licenses for Virgil C. Summer Nuclear Station, Units 2 and 3.

EDMGs

The NRC proposes to move the EDMGs requirement currently in § 50.54(hh)(2) to a new mitigation of beyond-design-basis events section of 10 CFR part 50. In addition to moving the text, the NRC proposes to make a few editorial changes. The wording used to describe these requirements has evolved from “guidance and strategies,” in Interim Compensatory Measures Order EA-02-026, dated February 25, 2002, to “strategies,” in the corresponding license conditions, to “guidance and strategies,” in § 50.54(hh)(2), to its proposed form “strategies and guidelines.” The word “guidelines” was chosen rather than “guidance” to better reflect the nature of the instructions that could be developed as appropriate by a licensee and to avoid confusion with the term “regulatory guidance.” The word “strategies” is used in the proposed rule to reflect its meaning, “plans of action.” The resulting plans of action could include plant procedures, methods, or other guideline documents, as deemed appropriate by the licensee during the development of these strategies. These plans of action would also include the arrangements made with offsite responders for support during an actual event. No substantive change to the requirements is intended by this proposed change in the wording.

Applicability of the requirements of § 50.54(hh)(2) is currently governed by § 50.54(hh)(3), which makes these requirements inapplicable following the submittal of the certifications required under § 50.82(a) or § 52.110(a)(1). As discussed in the statement of considerations for the Power Reactor Security Rulemaking (74 FR 13926), the NRC believes that it would be inappropriate for the requirements for EDMGs to apply to a permanently shutdown, defueled reactor, where the fuel was removed from the site or moved to an ISFSI. The NRC proposes to require EDMGs for a licensee with permanently shutdown defueled reactors, but with irradiated fuel still in its spent fuel pool, because the licensee must be able to implement effective mitigation measures for large fires and explosions that could impact the spent fuel pool while it contains irradiated fuel. The difference between the proposed rule and § 50.54(hh)(3) would correct the wording of the latter provision to implement the sunsetting of the associated requirement as was intended by the Commission in 2009. This change would not constitute backfitting for currently operating reactors because the proposed change concerns decommissioning reactors. The proposed change would not constitute backfitting for currently decommissioning reactors because the EDMGs are also required by the licensees' license conditions that were made generically applicable through the Power Reactor Security Rulemaking and remain in effect.

SAMGs

Nuclear plants are designed such that design basis events and accidents that result in challenges to normal or shutdown plant operations should not lead to significant damage to fuel in the reactor vessel or spent fuel pool. These design basis challenges are addressed through normal, abnormal, and emergency operating procedures, which are required by plant technical specifications. However, if the event or accident is outside the plant's design basis or the event or accident results in conditions in which the emergency operating procedures are unsuccessful in preventing the conditions in which imminent or actual significant fuel damage may occur, then the emergency operating procedures and other procedures or guidelines (e.g., EDMGs, AOPs (for spent fuel pool events, which were added to SAMGs following the Fukushima accident)) provide transition criteria to implement the SAMGs.

The NRC proposes to require strategies and guidelines for use by decision-making personnel under conditions in which significant damage to fuel in the reactor vessel or spent fuel pool is occurring or is imminent. Although there is not a standard definition for "significant damage" as related to fuel in the reactor vessel or spent fuel pool, in this context it means that fuel is damaged from inadequate cooling that results in fuel melting, such as what occurred in the TMI-2 accident and in the Fukushima Dai-ichi accidents. This condition is also referred to as a severe accident. These strategies and guidelines are commonly referred to as SAMGs. SAMGs are strategies and guidelines for use when entry conditions indicate the potential existence of an inadequate cooling condition for the reactor core or spent fuel pool that could lead to significant fuel damage. Entry conditions would likely be plant-specific, for example low reactor vessel water level in BWRs or high core exit temperatures for pressurized water reactors.

The proposed rule would require licensees to "maintain the strategies and guidelines," which means that licensees would be required to update their SAMGs consistent with the generic technical guideline documents applicable to their facility or ensure their site-specific SAMGs have been updated to incorporate relevant research and analysis and lessons learned from industry events. This also means that the SAMGs would have to be included in the configuration management program in order to ensure that the SAMGs reflect changes to the facility over time.

If a licensee chooses to develop, implement and maintain SAMGs independent of NRC-endorsed guidance, then the licensee should prepare a technical basis document supporting the strategies and guidelines incorporated into the SAMGs.

Integration with EOPs

In developing a proposed requirement for the integration of FSGs, EDMGs, and SAMGs with the EOPs, the NRC considered their differences in content and the standards for usage applied to them. The EOPs are a specific and prescribed set of instructions implemented in accordance with exacting standards for usage and adherence (e.g., step-by-step sequential performance, concurrent execution of multiple sections) that operators and plant staff are required to follow when performing a specific task or addressing plant conditions. When implementing procedures, each step is to be performed as prescribed, with rare exceptions. The strategies and guidelines that would be required differ from EOPs primarily in terms of the level of detail to which they are written and expectations regarding usage. These strategies and guidelines may be a less prescriptive set of instructions not subject to the same constraints imposed by standards of usage for procedure implementation (e.g., may not be followed in a step-by-step manner). This is because of: 1) the large number of possible event initiators, plant configurations, and sequences; and 2) the high degree of uncertainties in event progression and consequences. The strategies and guidelines can take the form of high level plans that identify and describe potential, previously evaluated, success paths for addressing specific conditions such as loss of core cooling. As a result, strategies and guidelines provide operators and plant staff the information and latitude to respond as necessary to unpredictable and dynamic situations, allowing them to adapt to the actual conditions and damage states without the burden of detailed procedures and the challenge of determining which procedure may be applicable and effective under the uncertain conditions of a beyond design basis accident.

Given these differences in content and standards for usage, the intent of the proposed rule is not to require conformance of the strategies and guidelines to the level of detail and standards of usage for EOPs, or consolidation of the strategies, guidelines and procedures into a single set of instructions, but rather, as previously described, to require functional integration of strategies and guidelines with the EOPs. The objective is for the strategies, procedures, and guidelines to retain or employ the characteristics that support their effective use under the range of conditions to which they are each intended to apply while ensuring that the strategies and guidelines, in conjunction with the EOPs, constitute a useable and cohesive set of instructions for mitigating the consequences of a wide range of initiating events and plant damage states. To achieve this functional integration, the NRC expects that applicants and licensees will have addressed the interfaces, dependencies, and interactions among the strategies and guidelines that would be required under the proposed rule and the EOPs, such that they can be implemented in concert with each other, as necessary, to effectively use available plant resources and direct a logical and coordinated response to a wide range of accident conditions.

In keeping with the basis for a functional integration of the strategies and guidelines with EOPs, the proposed rule would require that the FSGs, EDMGs, and SAMGs be integrated "with the Emergency Operating Procedures (EOPs)." This proposed language is intended to communicate the NRC's expectation that the EOPs retain their role as the primary means of directing emergency operations and that the strategies and guidelines that would be required under the proposed rule would be integrated with EOPs to support their implementation or augment them where their implementation is not successful in preventing significant fuel damage.

The NRC considered establishing specific criteria for the integration of the strategies and guidelines with EOPs but opted to specify only a high level requirement to allow applicants and licensees flexibility in the means by which they achieve the functional integration described previously. Common means that licensees and applicants will use to achieve functional integration might include the following:

1. Strategies, guidelines, and procedures have clearly defined transitions (e.g., entry and exit conditions with distinct pointers) from one strategy, guideline, or procedure to another.

2. Individuals are cued by the document or trained to know when transitions between the strategies, guidelines, and procedures result in corresponding changes in the associated standards for usage (e.g., when transitioning from EOPs to SAMGs, the operator is able to recognize the transition from a step-by-step procedure to a flexible guideline set where it is permissible to deviate from the order or method of accomplishing the steps).

3. Licensees establish expectations (e.g., through standards for usage) pertaining to the parallel use of strategies, guidelines, and procedures. Plant personnel using different strategies, guidelines, and procedures concurrently understand which is the controlling procedure and therefore which actions take precedence.

4. Licensees identify and resolve conflicts between the strategies, guidelines and procedures.

5. Licensees identify competing considerations when using the strategies, guidelines and procedures and eliminate or address them in guidance.

6. Licensees control the development and maintenance of their content and format in accordance with human factors standards and guidelines (e.g., writer’s guides) that recognize and address the interfaces between them in order to achieve compatibility of the strategies, guidelines, and procedures.

Staffing

The NRC proposes to require licensees to provide the staffing necessary for having an integrated response capability to support implementation of the FSGs, EDMGs, and SAMGs. To be effective, staffing for an expanded response capability should include the trained and qualified individuals who would be relied upon to analyze, recommend, authorize, and implement the mitigating strategies. The staffing must directly support the assessment and implementation of a range of mitigation strategies intended to maintain or restore the functions of core cooling, containment, and spent fuel pool cooling.

The staffing analyses required by proposed appendix E, section VII should determine when personnel performing expanded response functions should report to the site, within a timeframe sufficient to support implementation of the strategies that are not assigned to the on-shift staff. This will ensure that the functions of core cooling, containment, and spent fuel pool cooling are continuously maintained or are promptly restored.

The NRC has endorsed the industry guidance for conducting staffing analyses, NEI 10-05, “Assessment of On-Shift Emergency Response Organization Staffing and Capabilities,” Revision 0, and NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” Revision 0, and the NRC has issued Interim Staff Guidance (ISG), NSIR/DPR-ISG-01, “Emergency Planning for Nuclear Power Plants,” that provides the requisite details in determining the staffing levels and for which positions, as well as, which beyond design basis external events the applicants and licensees should evaluate.

The recommended minimum positions and staffing levels for emergency plans were initially provided in NUREG-0654/FEMA-REP-1, Revision 1, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants.” Following the September 11, 2001 events, the NRC issued Enhancements to

Emergency Preparedness Regulations (EP final rule) (76 FR 72560) to amend 10 CFR part 50, appendix E, to address, in part, concerns about the assignment of tasks or responsibilities to on-shift emergency response organization (ERO) personnel that would potentially overburden them and prevent the timely performance of their functions under the emergency plan. Licensees must have enough on-shift staff to perform specified tasks in various functional areas of emergency response 24 hours a day, 7 days a week. The proposed rule addresses the staffing requirements for the expanded response capabilities for on-shift response and the ERO.

The proposed rule would require adequate staffing to implement the FSGs, EDMGs, and SAMGs with the EOPs without requiring further analysis to supplement analyses that were completed as a result of post-Fukushima orders or the EP final rule. Staffing levels should be established to ensure that if strategies are executed there would be no delays in completing them caused by the lack of qualified personnel. The NRC expects that the use of drills, existing training analyses and other methods would verify sufficient staffing levels.

Command and Control

The NRC proposes to require licensees to have a supporting organizational structure with defined roles, responsibilities, and authorities for directing and performing the FSGs, EDMGs, and SAMGs. The objective is to ensure that licensees address the organizational implications of: (1) implementing the FSGs; and (2) integrating the FSGs, EDMGs, and SAMGs with the EOPs such that organizational units responsible for on-site accident mitigation (e.g., main control room, emergency operations facility, and technical support center staff) can support a coordinated implementation of these procedures and guidelines under the challenging conditions presented by beyond-design-basis events.

Additional requirements currently exist in 10 CFR part 50, appendix E, section IV.A for the inclusion within the emergency plan of a description of the organization for coping with radiological emergencies, including definition of authorities, responsibilities, and duties of individuals assigned to the licensee's emergency organization and the means for notification of such individuals in the event of an emergency. These requirements provide the command and control structure for use in the execution of the emergency plan. The current 10 CFR part 50, appendix E, section IV.A.2.a. and 5., further require that the emergency plan include 1) a detailed description of the authorities, responsibilities, and duties of the individual(s) who will take charge during an emergency; 2) plant staff emergency assignments, authorities, responsibilities, and duties of an onsite emergency coordinator who shall be in charge of the exchange of information with offsite authorities responsible for coordinating and implementing offsite emergency measures; and 3) the identification, by position and function to be performed, of other employees of the licensee with special qualifications for coping with emergency conditions that may arise.

The need for defined command and control structures and responsibilities for use in beyond-design-basis conditions was recognized in the course of the development of the guidance and strategies for the current § 50.54(hh)(2). As stated in the industry's guidance document for that set of requirements, NEI 06-12, "B.5.b Phase 2 & 3 Submittal Guideline," Revision 2, "Experience with large scale incidents has shown that command and control execution can be a key factor to mitigation success." The guidance and strategies developed for that effort include an EDMG for initial response to provide a bridge between normal operational command and control and the command and control that is provided by the ERO in the event that the normal command and control structure is disabled. The NRC considers that the actions taken in the development of the EDMG for initial response for the guidance and strategies for the current § 50.54(hh)(2) would continue to be adequate for compliance with the proposed rule for EDMGs following the proposed movement of those requirements.

The endorsed industry guidance in NEI 12-06, Revision 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," for the guidance and strategies required by Order EA-12-049, specifies that the existing command and control structure will be used for transition to the SAMGs.

All previous requirements did not specify a command and control structure for a multi-unit event that includes the potential need for acquisition of offsite assistance to support onsite event mitigation. Additionally, these requirements were not understood to require such a response since they preceded the Fukushima event and the regulatory actions that stemmed from that event. As a practical matter, the current command and control structures, including any changes that resulted from the implementation of Order EA-12-049 requirements, are expected to be sufficient to ensure that the functional objectives of the proposed rule are achieved. Accordingly the NRC recognizes that this new requirement may not be necessary and is requesting stakeholder feedback on this issue (refer to section VII of this notice).

Equipment

The NRC proposes to have requirements for licensee equipment, including instrumentation, that is relied upon for use in the proposed mitigation strategies and guidelines. This rulemaking does not propose to modify the regulatory treatment of equipment relied upon for the EDMGs currently required by § 50.54(hh)(2). The regulatory treatment of that equipment will remain as it is described in the endorsed guidance document for those strategies and guidelines. This rulemaking also proposes no modifications to the regulatory treatment of equipment relied upon for the proposed SAMGs. This is because no additional equipment requirements are contemplated by this rulemaking, which is limited to the development, implementation, and maintenance of SAMGs as a decision-making framework for choosing between pre-planned responses relying on existing equipment. 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, EPRI has developed a technical basis document, the TBR, 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 proposed rule would make generically applicable requirement (2) of Order EA-12-049, attachments 2 and 3, which reads as follows: “These strategies must … have adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this Order.”

The industry guidance of NEI 12-06, as endorsed by NRC interim staff guidance JLD-ISG-2012-01, “Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events,” included specifications for licensee provision of a spare capability in order to assure the reliability and availability of the equipment required to provide the capacity and capability requirements of the Order. This spare capability was also referred to within the guidance as an "N+1" capability, where "N" is the number of power reactor units on a site. The NRC considered including requirements similar to the spare capability specification of NEI 12-06 in the proposed rule but determined that such an inclusion would be too prescriptive and could result in the need to grant exemptions for alternate approaches that provide an effective and efficient means to provide the required capability of the Order. One example of this is in the area of flexible hoses for which a strict application of the sparing guidance could necessitate provision of spare hose or cable lengths sufficient to replace the longest run of hoses when significant operating experience with similar hoses for fire protection does not show a failure rate that would support this as a need.

The development of the mitigating strategies in response to Order EA-12-049 relied upon a variety of initial and boundary conditions that were provided in the regulatory guidance of JLD-ISG-2012-01, Revision 0, and NEI 12-06, Revision 0. These initial and boundary conditions followed the philosophy of the basis for imposition of the requirements of Order EA-12-049, which was to require additional defense-in-depth measures to provide continued reasonable assurance of adequate protection of public health and safety. As a result, the industry response to Order EA-12-049 includes diverse and flexible means of accomplishing safety functions rather than providing an additional further hardened train of safety equipment. These requirements and conditions included the acknowledgement that, due to the fact that initiation of an event requiring use of the strategies would include multiple failures of safety-related structures, systems, and components (SSCs), it is inappropriate to postulate further failures that are not consequential to the initiating event. As a result, the NRC has determined that the conditions to which the instrumentation relied on for the mitigating strategies would be exposed do not include conditions stemming from fuel damage, but instead are limited as described previously. The NRC has determined that it should not be necessary for the instrumentation to be designed specifically for use in the mitigating strategies and guidelines, but instead it would be necessary that the design and associated functional performance be sufficient to meet the demands of those strategies.

The underlying proposed requirements are for events that are not included in the design basis events as that term is used in the § 50.2 definition of safety-related SSCs. Because of this, reliance on equipment for use in the related strategies would not result in the applicability of 10 CFR part 50, appendix A, General Design Criterion (GDC)-2, “Design bases for protection against natural phenomena,” or the principal design criterion (PDC) applicable to a plant’s operating license if issued prior to GDC-2. The proposed rule would require reasonable protection for the equipment relied on for the mitigation strategies to a hazard level as severe as that originally determined for the facility under GDC-2 or the applicable PDC unless the reevaluated hazards stemming from the March 12, 2012, NRC letter issued under § 50.54(f), as assessed by the NRC show that increased protection is necessary. The March 12, 2012, NRC letter requested information on licensees’ seismic and flooding hazards; licensees and the NRC are currently scheduled to complete most of the work on the flooding reevaluations prior to the anticipated effective date of this proposed rule. The NRC notes that there are some licensees whose licensing bases include requirements for protection from natural phenomena beyond those established at the original licensing (e.g., North Anna Power Station for the seismic hazard), but anticipates that these different hazard levels will be captured in the reevaluation of external hazards under the March 12, 2012, NRC letter.

As discussed in COMSECY-14-0037, “Integration of Mitigating Strategies for Beyond-Design-Basis External Events and The Reevaluation of Flooding Hazards,” (ADAMS Accession No. ML14309A256) and its associated SRM, the requirements of Order EA-12-049 were imposed in parallel with the agency’s March 12, 2012, requests for information on the reevaluation of external hazards. As a result, Order EA-12-049 included a requirement in both

Attachment 2 and 3 for licensees to provide reasonable protection for equipment associated with the required mitigating strategies from external events without specific reference to the necessary level of protection. The appropriate level of protection from external hazards, particularly flooding, was the subject of discussion in the course of NRC-held public meetings leading up to the issuance of JLD-ISG-2012-01 and its endorsement of the industry guidance for Order EA-12-049, NEI 12-06. Section 6.2.3.1 of NEI 12-06 specifies that the level of protection for flooding should be "the flood elevation from the most recent site flood analysis. The evaluation to determine the elevation for storage should be informed by flood analysis applicable to the site from early site permits, combined license applications, and/or contiguous licensed sites." The choice of this hazard level was driven by the recognition that while the flooding hazard reevaluations by holders of operating licenses and construction permits may not be complete in advance of the development and implementation of the mitigating strategies, but that information available from flood analyses for nearby sites could be taken into account in choosing the appropriate level in order to avoid the need for rework or modification of the strategies. Many licensees took the former approach, using their best estimates of potential hazard levels and providing additional margin to the current licensing basis. (See, e.g., the description of the flooding strategies for Fort Calhoun Station on page B-43 et seq., of Omaha Public Power District's Overall Integrated Plan (Redacted) in Response to March 12, 2012, Order EA-12-049.)

In COMSECY-14-0037, the NRC staff requested that the Commission affirm that: 1) licensees for operating nuclear power plants need to address the reevaluated flooding hazards within their mitigating strategies for beyond-design-basis external events; 2) licensees for operating nuclear power plants may need to address some specific flooding scenarios that could significantly damage the power plant site by developing targeted or scenario-specific mitigating strategies, possibly including unconventional measures, to prevent fuel damage in reactor cores or spent fuel pools; and 3) the NRC staff should revise the flooding assessments and integrate the decision-making into the development and implementation of mitigating strategies in accordance with Order EA-12-049 and this rulemaking. These principles reflect the NEI 12-06 reference to the "most recent flood analysis" discussed above and the documentation by licensees in their overall integrated plans for the mitigating strategies that at the time of their submittals "flood and seismic reevaluations pursuant to the § 50.54(f) letter of March 12, 2012, are not completed and therefore not assumed in this submittal. As the reevaluations are completed, appropriate issues will be entered into the corrective action system and addressed on a schedule commensurate with other licensing bases changes." In SRM-COMSECY-14-0037, the Commission approved the first two items recommended by the NRC staff, regarding the need for operating nuclear power plant licensees to address the reevaluated flood hazards within the mitigating strategies and the potential for using targeted or scenario specific mitigating strategies. The Commission did not approve the third recommendation, but that recommendation is outside the scope of this rulemaking effort. The NRC drafted the proposed rule to reflect this direction and in recognition of the fact that the wording of Order EA-12-049 and its associated guidance did not make clear that the mitigating strategies equipment would require protection to the reevaluated hazard levels resulting from the § 50.54(f) request for information of March 12, 2012.

Because the events for which the proposed mitigating strategies are to be used are outside the scope of the design basis events considered in establishing the basis for the design of the facility, equipment that is relied upon for those mitigating strategies may not fall within the scope of § 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." Nevertheless, the NRC proposes that such equipment should receive adequate maintenance in order to assure that it is capable of fulfilling its intended function when called upon.

The NRC proposes to require licensees to have a means to remotely monitor wide-range SFP level as a part of the equipment relied upon to support the FSGs. This provision would make generically-applicable the requirements imposed by Order EA-12-051. The NRC considered including the detailed requirements from Order EA-12-051 within the proposed rule, but determined that the more performance-based approach taken with the proposed rule would better enable an applicant for a new reactor license or design certification to provide innovative solutions to address the need to effectively prioritize event mitigation and recovery actions between the source term contained in the reactor vessel and that contained within the spent fuel pool.

Design Features

The proposed rule would require that certain applicants include design features in the plant design sufficient to enhance coping durations and minimize reliance on human actions to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities during an ELAP concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink.

Order EA-12-049 was issued to all operating reactor licensees, construction permit holders, and the Vogtle Electric Generating Plant, Units 3 and 4 licensee. This order requires licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities (i.e., key safety functions) following a beyond-design-basis external event. Based on the NRC's assessment of new insights from the events at Fukushima Dai-ichi, it concluded that these additional requirements must be imposed on licensees or construction permit holders to increase the capability of nuclear power plants to mitigate beyond-design-basis external events, and that these additional requirements are needed to provide adequate protection to public health and safety. The three-phase approach described in the order is a conceptual framework built upon the need for a licensee to address challenges to key safety functions using installed SSCs until portable equipment is available to address those challenges. Current NRC-endorsed guidance (e.g., NEI 12-06) has focused on operating reactors and reflects that operating reactors are constrained by existing SSCs and plant layouts. However, without such constraints, new reactor applicants have an opportunity to incorporate enhanced capability for mitigating strategies into the design.

New reactor applicants can address major elements of Order EA-12-049 by incorporating into the plant design those design features for mitigating strategies that provide enhanced capability to ensure that key safety functions are maintained or restored. Such design features should reduce reliance on and simplify the manual actions necessary to maintain or restore key safety functions, and allow more time to assess plant conditions and prolong the use of installed equipment, as compared to currently operating reactors. This approach is consistent with the Policy Statement on the Regulation of Advanced Reactors (73 FR 60612; October 14, 2008), in which the Commission encouraged vendors to include certain features into the plant design; and the most effective opportunity to do so is during the design of those SSCs.

During the design of the SSCs that perform key safety functions, an applicant would have an opportunity to incorporate additional capability to mitigate beyond-design-basis external events. That is, rather than relying heavily on portable (flexible) equipment to mitigate these events as with currently operating reactors, new reactor applicants would be able to reduce or delay reliance on portable equipment and offsite resources through engineered design features. New reactor applicants would also have greater flexibility to include these design features at the design stage without the constraints on operating reactors. As a result, design features for new reactor designs would enhance the mitigation strategies by increasing the capability, reliability, and flexibility of the SSCs relied upon for the mitigating strategies.

The regulatory approach of establishing alternative design requirements for new reactor applicants is consistent with the NRC's past practice. For example, § 50.150, "Aircraft Impact Assessment," requires new reactor applicants, in part, to perform an assessment and incorporate into the design capabilities to assure core cooling or containment integrity, and spent fuel pool cooling or integrity.

The Commission also has directed that new reactors incorporate certain safety enhancements that provide additional safety margin and defense-in-depth recognizing the opportunities that are available during the design stage. For example, in SRM-SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated June 26, 1990, the Commission approved the NRC staff's position that new reactors address § 50.63 through installation of an alternate ac source of diverse design capable of powering at least one complete set of normal shutdown loads, as opposed to extended coping capabilities. Additionally, in the same SRM, the Commission approved the NRC staff's position that new reactors provide physical separation for fire protection equipment to ensure that safe shutdown can be achieved, assuming the loss of a fire area, rather than rely upon fire barriers, to meet § 50.48, "Fire Protection."

The objective of the design features section of the proposed rule is to require that new nuclear power reactor applicants include in the facility design the capability to mitigate the effects of an ELAP. Such an event could be caused by an external hazard (e.g., a flood) and require mitigation during the period when conditions created by the external event could hinder or prevent access to protected on-site portable equipment and associated connections. Applicants would be required to incorporate design features to maintain or restore key safety functions. Extreme hazard events and the resulting effects on regional infrastructure (which may complicate operations) can last for several days as experienced at the Fukushima Dai-ichi nuclear plant site during the Great East Japan Earthquake and subsequent tsunami in 2011, the Turkey Point nuclear plant site during hurricane Andrew in 1992, and during hurricane Katrina in 2005. Although the effects can last for several days, they should not preclude access to staged equipment. The design features should result in a plant design that is better equipped for an ELAP condition and thus have a longer period of time until the facility would need to rely on portable equipment and offsite resources to maintain or restore key safety functions.

The benefit of being able to cope with an ELAP with reduced reliance on operator actions was recognized by the NTTF Report based on the review of insights from the Fukushima Dai-ichi accident. Greater reliance on design features that include well thought-out human-machine interfaces would reduce reliance on and simplify the manual actions necessary to maintain or restore key safety functions. Further, reducing reliance on human actions would also reduce the potential for human failures during stressful, adverse conditions.

While the requirement to incorporate design features would result in an improved capability to mitigate an ELAP, which could be caused by a beyond-design-basis external event, the NRC does not intend for a licensee to rely solely on these design features. In light of the unknown severity, duration, or other aspects of beyond-design-basis events that could result in an ELAP, it is important for a licensee to have diverse, flexible strategies for addressing such events that include design features as one aspect of those strategies. In other words, including these design features would not preclude the need for portable equipment and offsite resources in a licensee's mitigation strategies. Flexible strategies would also allow for contingencies if the design features are not available or otherwise not performing as expected. Therefore, while the designer of key safety systems would be required to incorporate design features to better enable the licensee to mitigate these events, a licensee would need to ensure its mitigation strategies also include diversity and flexibility through portable equipment and offsite resources.

The NRC is proposing to require these design features only for applicants for new nuclear power reactor designs. These design features would be included when the SSCs that support key safety functions are being designed (such as during the development of a design certification application) and not when including such design features could impose a significant burden on licensees or applicants and potentially decrease standardization (such as a combined license application referencing a certified design). The NRC is not proposing to require licensed nuclear power plants, certified designs, or license applications referencing an approved design (e.g., a design certification) to incorporate such design features because the potentially significant modifications to the design have not been justified under the NRC's backfit rule at 10 CFR 50.109 or the issue finality provisions of 10 CFR part 52.

As a result, the NRC is proposing that new nuclear power reactor designs should be capable of maintaining or restoring key safety functions of core cooling, containment, and spent fuel pool cooling capabilities using design features (i.e., installed SSCs) for near-term coping for as long as practical following initiation of an ELAP event. The extended coping duration would provide a greater amount of time to plan and deploy portable on-site equipment for long-term coping.

An NRC staff member did not agree with proposing requirements for new reactor designs to incorporate design features as part of this rule and submitted a non-concurrence. In accordance with the NRC's non-concurrence process, NRC management and staff considered, and responded to, the staff member's concerns, and documentation of the non-concurrence can be found in "Non-concurrence NCP-2015-003."

Training

The NRC anticipates that mitigation of the effects of beyond-design-basis events using the proposed strategies and guidelines would be principally accomplished through manual actions rather than automated plant responses for currently operating reactors. New reactors would be less reliant on manual actions due to incorporation of design features simplifying the need for operator actions and providing longer time periods prior to any needed actions. Additionally, the NRC notes that it anticipates that instructions provided for event mitigation may be largely provided as high level strategies and guidelines rather than step-by-step procedures. The use of strategies and guidelines supports the ability to adapt the mitigation measures to the specific plant damage and operational conditions presented by the event. However, effective use of this flexibility will depend upon the knowledge and abilities of personnel to select appropriate strategies or guidelines from a range of options and implement mitigation measures using equipment or methods that may differ from those employed for normal operation or design-basis event response. As a result, the NRC considers personnel training and qualification necessary to ensure that individuals will be capable of effectively performing their roles and responsibilities in accordance with the strategies and guidelines that would be required by this proposed rule.

The NRC acknowledges that licensee training programs, such as those required for licensed operators under 10 CFR part 55, “Operators’ Licenses,” the programs for plant personnel specified under § 50.120, "Training and Qualification of Nuclear Power Plant Personnel,” and the training for emergency response personnel required by 10 CFR part 50, appendix E, section IV.F, “Training,” would likely provide for many of the knowledge and abilities required for performing activities in accordance with the strategies and guidelines that would be required by this proposed rule. Nevertheless, as noted previously, the NRC anticipates that these strategies and guidelines may use new methods or equipment that require knowledge and abilities not currently addressed under existing training programs and as a result there may be gaps in these training programs that must be addressed to support effective use of the strategies and guidelines. Accordingly, this proposed rule would further require that licensees provide for the training of personnel using a systems approach to training as defined in § 55.4 (the Systems Approach to Training (SAT) process), except for elements already covered under other NRC regulations[6]. The SAT process, which is acceptable for meeting training requirements under 10 CFR part 55 and § 50.120, would also be appropriate for licensee identification and resolution of any current gaps or future modifications to personnel training that may be necessary to provide for the training of personnel performing activities in accordance with the mitigating strategies and guidelines that would be required by this proposed rule. The NRC recognizes that there are other training programs that are currently acceptable for meeting other regulatory required training (e.g., 10 CFR part 50, appendix E, section IV.F) that do not use the SAT process. In light of the existence of these training programs, which have been found acceptable for more frequently occurring design-basis events, the NRC has determined that these training programs can meet the needs for common elements with beyond-design-basis event mitigation. Therefore, the NRC would not require licensees to revise these training programs to use the SAT process to meet the proposed requirements. Licensees would be required to use the SAT process for newly identified training requirements supporting the effective use of the strategies and guidelines that would be required by this proposed rule.

By using the SAT process, licensees would identify and train on any additional tasks that would be necessary to implement the strategies and guidelines for the mitigation of beyond-design-basis events as defined in this proposed rule. The additional tasks identified would be incorporated into the training program to ensure appropriate training would be administered for each qualified individual designated to implement the strategies and guidelines required by this proposed rule.

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[6] This definition of a systems approach to training (SAT), is a training program that includes the following five elements: 1) systematic analysis of the jobs to be performed; 2) learning objectives derived from the analysis which describe desired performance after training; 3) training design and implementation based on the learning objectives; 4) evaluation of trainee mastery of the objectives during training; and 5) evaluation and revision of the training based on the performance of trained personnel in the job setting.

The new training needs that result from identification of any new tasks should be evaluated by licensees to determine whether each of their plant referenced simulators can be used to allow personnel to simulate their actions and resulting plant response. The NRC would not require licensees to update their plant referenced simulators to model conditions that may exist during significant damage to fuel in the reactor vessel or spent fuel pool. However, if a licensee's plant referenced simulator is capable of modeling these conditions (i.e., has an engineering basis to support such a conclusion) then use of the simulator capability would be an effective means to support the licensee's training designed to implement the mitigating strategies and guidelines required by this proposed rule.

The NRC originally proposed a possible revision to §§ 55.41, 55.43, and 55.45, which was discussed in the Onsite Emergency Response Capabilities rulemaking regulatory basis. The proposed changes would have added requirements that initial licensing examinations, both the written and operating exams, for operators and senior operators would include SAMGs as a required topic for training. During several public meetings the nuclear power industry strongly voiced the concern that this would raise the level of training for SAMGs equal to the level of training for the EOPs, and the industry proposed that NRC consider revising the training requirement to be consistent with the SAT process. The proposed revision would allow the SAT process to determine the extent of the training for SAMGs and would not result in the displacement of higher priority training topics, such as EOPs. Additionally, the industry stated that revising §§ 55.41, 55.43, and 55.45 to include SAMGs as required training for newly licensed operators would negatively impact the overall length of the operator licensing training program, which is currently an average of 22 months in length. The NRC is proposing to require training based on the SAT process instead of revising §§ 55.41, 55.43, and 55.45 because the SAT process is well-established and required for initial licensing of operators under 10 CFR part 55 and to train and qualify individuals identified in § 50.120.

Consistent with this proposed approach, the NRC has proposed a change to the current knowledge and abilities catalogs that are used to develop initial operator licensing examinations for operating power reactors to include additional knowledge and abilities that could be tested on the initial operator licensing written examinations and operating tests. Because the initial licensing written examinations are developed based on items randomly selected from the knowledge and abilities catalogs for each vendor type of reactor, the NRC proposes to add two additional knowledge and abilities related to severe accidents that could be selected for the written examination and portions of the operating test. These new knowledge and abilities are related to: 1) the transitions from the EOPs to the SAMGs; and 2) the lines of authority during implementation of the SAMGs. These new knowledge and abilities are consistent with the newly developed knowledge and abilities catalog that is in place for the Advanced Boiling Water Reactor plant design.

Change Control

The proposed requirements address beyond-design-basis events, and as such, currently existing change control processes do not address all aspects of a contemplated change, including most notably § 50.59. As such, the proposed change control provision is intended to supplement the existing change control processes and focus on the beyond-design-basis aspects of the proposed change.

This proposed rule would not contain criteria typically included in other change control processes that are used as a threshold for determining when a licensee needs to seek NRC review and approval prior to implementing the proposed change. Instead, the proposed provisions would require that the evaluations of the proposed change reach a conclusion that all new requirements continue to be met and that this evaluation is documented and maintained to support NRC inspection.

Proposed changes that remain consistent with regulatory guidance would be acceptable, since such changes would ensure continued compliance with the proposed provisions in this rulemaking. The NRC recognizes that the proposed change control provisions may result in licensees seeking NRC review and approval of proposed changes that do not follow current regulatory guidance for this proposed rulemaking potentially through a license amendment or through NRC review of new or revised regulatory guidance. Accordingly, the NRC is requesting stakeholder feedback on this issue to determine whether there is a better regulatory approach for change control (refer to the “Specific Requests for Comments” section of this document).

During public discussions before issuance of this proposed rule, there was a suggestion that the NRC should consider a provision to allow a licensee to request NRC review of a proposed change, and that if the NRC did not act upon the request for a suggested time period (e.g., 180 days) that the request be considered “acceptable.” The NRC did not include this “negative consent” type of approval process in the proposed rule and instead the proposed change control process places the responsibility on the licensees to ensure that proposed changes result in continued compliance with the proposed rule provisions, or are otherwise submitted to the NRC following the § 50.12 exemption process. The NRC expects to obtain stakeholder feedback on this issue and will consider that feedback when developing the final rule provisions.

A licensee may intend to change its facility, procedures, or guideline sets to revise some aspect of beyond design basis mitigation (i.e., governed by the proposed provisions of this rulemaking), and the same change can impact multiple aspects of the facility (i.e., impact “design basis” aspects of the facility and be subject to other regulations and change control processes). As previously discussed, the NRC anticipates that a licensee would ensure that a proposed changed is consistent with endorsed guidance to ensure continued compliance with the proposed provisions. This same change could also impact safety-related structures, systems, and components, either directly (e.g., a proposed change that impacts a physical connection of mitigation strategies equipment to a safety-related component or system) or indirectly (e.g., a proposed change that involves the physical location of mitigation equipment in the vicinity of safety-related equipment that presents a potential for adverse physical/spatial interactions with safety-related components). As such, § 50.59 would need to be applied to evaluate the proposed change for any potential impacts to safety-related SSCs.

Additionally, proposed changes can impact numerous aspects of the facility beyond the safety-related impacts, including implementation of fire protection requirements, security requirements, emergency preparedness requirements, or safety/security interface requirements. Accordingly, it would be necessary for a licensee to ensure that all applicable change control provisions are used to judge the acceptability of facility changes including, for example, change control requirements for fire protection, security, and emergency preparedness. Additionally, recognizing the nature of mitigation strategies and the reliance on human actions, it is also necessary to ensure that the proposed changes satisfy the safety/security interface requirements of § 73.58. It is the obligation of the licensee to comply with all applicable requirements, and as such, the proposed change control provisions could be viewed as unnecessary. However recognizing the potential complexity of proposed facility changes and the complexity of existing regulatory requirements that govern change control, the NRC concluded that adding the proposed change control provision, for the purposes of regulatory clarity, was warranted.

As another example of change control processes other than those proposed in this rulemaking, section VIII of each design certification rule in the appendices of 10 CFR part 52 includes a process for changes to and departures from the design control document. If a new certified design, including a similar change control process, were to be subject to the proposed design features requirement, and a licensee referencing that certified design proposed changes to the design features under the proposed design features requirement, the licensee would follow the requirements of section VIII of that design certification rule rather than the proposed change control process. The NRC plans to add a new change control provision to future design certification rules subject to the proposed design features requirement (including amendments to any of the five existing design certifications) to govern combined license applicants and holders referencing the design certification that request a departure from the design features in the referenced design certification. The new change control provision would require that, if the applicant or licensee changes the information required by § 52.47(a)(29) to be included in the final safety analysis report (FSAR) for the standard design certification, then the applicant or licensee shall ensure that the changed feature continues to satisfy the design objectives described in the proposed design features requirement .

Implementation

The NRC proposes a compliance schedule of four years following the effective date of the rule. This proposed rule does not include any special provision for a holder of a COL as of the effective date of the rule for which the Commission has not made the finding required under § 52.103(g) (i.e., a COL holder still in the construction phase). The NRC considers the duration of four years prior to compliance with the requirements of the proposed rule to be acceptable because the majority of these requirements have been previously implemented under Orders EA-12-049 and Order EA-12-051 or § 50.54(hh)(2), or are in response to the § 50.54(f) requests for information issued March 12, 2012. For the proposed SAMG requirements, substantial work has already been completed as a result of the previous voluntary industry initiative, and as a result of the recent efforts on the part of industry to update the generic SAMG guidance.

Regulatory Basis for New Emergency Response Capability Requirements

A significant objective of this rulemaking is to make the requirements that were previously imposed under Order EA-12-049 generically applicable. As an implicit part of the implementation of Order EA-12-049, additional emergency response capabilities were included to address a beyond-design-basis external event that impacts multiple power reactor units, and potentially multiple source terms, on the site. In all cases these additional proposed revisions are considered to be necessary to effectively mitigate such an event, consistent with the NRC's intent in issuance of Order EA-12-049. These proposed requirements were not explicitly addressed in the previous regulatory bases documents issued for the two rulemakings. This section discusses the basis for these proposed emergency response capability provisions.

The March 12, 2012, § 50.54(f) letters (i.e., Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f)) requested information from the licensees that, in part, was intended to verify the adequacy of emergency planning to address what was then termed prolonged SBO[7] and multi-unit events. The accident at Fukushima highlighted the need to determine and implement the required staff to fill all necessary positions responding to multi-unit events. Additionally, NRC recognizes that the communication equipment relied upon to coordinate the event response during an ELAP should be powered and maintained.

1. Onsite and offsite communications capability

This proposed rule would require additional communications capabilities for events that result in extended loss of ac power onsite, or potential destruction of offsite communications infrastructure. Because of the destruction to communications capability that occurred at Fukushima, the NRC would propose requirements for licensees to provide a greater capability

[7] While the letter made use of the term "prolonged SBO," the request for information was for a loss of all alternating current power, which was subsequently termed an ELAP. The phrase "prolonged SBO" is retained here to accurately reflect the wording used in the letter.

to communicate with onsite staff to support mitigation of the event, and to support offsite communications to gain any additional support or to perform emergency preparedness functions. The proposed requirements would support effective implementation of the FSGs and were included as part of the implementation of Order EA-12-049.

2. Staffing assessment

This proposed rule would require an assessment that is considered essential to effective implementation of the FSGs. This assessment matches the one that was conducted under the March 12, 2012, request for information that was developed to align with the requirements included in Order EA-12-049 (i.e., the staffing analysis specifically considered the staffing needs for implementing Order EA-12-049); licensees would not be required to repeat the staffing analysis. A lesson-learned from the Fukushima event is that there are increased staffing demands following a beyond-design-basis external event, and this coupled with the subsequent NRC requirements issued in Order EA-12-049 required the staffing analysis to provide a level of assurance that the FSGs can be implemented. This provision would then support the proposed requirements of the rule to have sufficient staffing to implement the FSGs, EDMGs, and SAMGs in conjunction with the EOPs.

3. Change control

The NRC would not require a power reactor applicant or licensee to address or implement the proposed communications and staffing analysis requirements through the licensee's or applicant's emergency plan or maintain the capabilities as a part of the emergency preparedness program. This approach would allow for site-specific flexibility in implementation. Therefore, the requirements of maintaining the communications and staffing analysis in an effective emergency plan and controlling changes to it under § 50.54(q) would not apply when implementation of the requirements is not in the emergency plan, but in all cases, the change control process of the proposed rule would apply. However, if an applicant or a licensee incorporates the communications and staffing analysis into the emergency preparedness program through the emergency plan or emergency plan implementing procedures, the requirements of § 50.54(q) would apply.

4. Multi-source term assessment capability

This proposed rule would require licensees to have a means for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials, including from all reactor core and spent fuel pool sources. A lesson learned from the Fukushima Dai-ichi event is that there is a potential for a beyond-design-basis external event to result in multiple source terms from multiple release points, and under such a situation, additional capabilities are necessary to support development of appropriate protective action recommendations. In COMSECY-13-0010, “Schedule and Plans for Tier 2 Order on Emergency Preparedness for Japan Lessons Learned,” dated March 27, 2013, the NRC staff informed the Commission that licensees would provide information about their current multiple source assessment capability, or a schedule for implementing such a capability, and that associated implementation would occur by the end of calendar year 2014. Licensee implementation of the multiple source assessment capability will be verified by inspection under TI-2515/191, “Inspection of the Licensee's Responses to Mitigation Strategies Order EA-12-049, Spent Fuel Pool Instrumentation Order EA-12-051 and Emergency Preparedness Information Requested in NRC March 12, 2012.” The NRC has been working with the industry and stakeholders through public meetings to review and provide feedback on NEI 13-06, “Enhancements to Emergency Response Capabilities for Beyond Design Basis Accidents and Events,” Revision 0, which, in part, would provide licensees with guidance on implementing a multiple source dose assessment capability.

The capability should be available to support responses during events both within and beyond the plant design basis. Also, the licensee should discuss the site’s multi-unit and multiple source dose assessment capability with the offsite response organizations, particularly, with the agencies that are responsible for making decisions on public protective action recommendations. Agreement on the methods and results will avoid unnecessary delays during the event in making the public protective action decisions, public notification, and the implementation of protective actions.

5. Technology-neutral Emergency Response Data System

The proposed requirements of 10 CFR part 50, appendix E, section VI, for the Emergency Response Data System (ERDS) would reflect the use of up-to-date technologies and remain technology-neutral so that the equipment supplied by NRC would continue to be replaced as needed, without the need for future rulemaking because equipment becomes obsolete. In 2005, the NRC initiated a comprehensive, multi-year effort to modernize all aspects of the ERDS, including the hardware and software that constitute the ERDS infrastructure at NRC headquarters, as well as the technology used to transmit data from licensed power reactor facilities. As described in NRC Regulatory Issue Summary 2009-13, "Emergency Response Data System Upgrade From Modem to Virtual Private Network Appliance," the NRC engaged licensees in a program that replaced the existing modems used to transmit ERDS data with Virtual Private Network (VPN) devices. The licensees now have less burdensome testing requirements, faster data transmission rates, and increased system security.

V. Section-by-Section Analysis.

Proposed § 50.8 Information Collection Requirements: OMB Approval

This section, which lists all information collections in 10 CFR part 50 that have been approved by the Office of Management and Budget (OMB), is revised by adding a reference to § 50.155, the mitigation of beyond-design-basis events rule. As discussed in the "Paperwork Reduction Act Statement" section of this document, the OMB has approved the information collection and reporting requirements in the final mitigation of beyond-design-basis events rule. No specific requirement or prohibition is imposed on applicants or licensees in this section.

Proposed § 50.34 Contents of Applications; Technical Information

Section 50.34 identifies the technical information that must be provided in applications for construction permits and operating licenses. Paragraphs (a) and (b) of this section identify the information to be submitted as part of the preliminary or final safety analysis report, respectively. New paragraph (i) of this section would identify information to be submitted as part of an operating license application, but not necessarily included in the final safety analysis report.

The NRC is proposing an administrative change to § 50.34(a)(13) and (b)(12) to remove the word "stationary" from the requirement for power reactor applicants who apply for a construction permit or operating license, respectively. Section 50.34(a)(13) and 50.34(b)(12) were added to the regulations in 2009 to reflect the requirements of § 50.150(b) regarding the inclusion of information within the preliminary or final safety analysis reports for applicants subject to § 50.150. Section 50.34(a)(13) and (b)(12) were inadvertently limited to "stationary power reactors," matching the wording of § 50.34(a)(1), (a)(12), (b)(10), and (b)(11), which pertain to seismic risk hazards for stationary power reactors. The NRC does not intend to change the meaning of this requirement by removing the word "stationary" from these requirements. This change is intended to ensure consistency in describing the types of applications to which the requirements apply.

Proposed § 50.34(a)(14) would require applicants for a construction permit to submit a preliminary description of the design features included in the plant design in the preliminary safety analysis report under proposed § 50.155(d), and an explanation of how those design features comply with the requirements of proposed § 50.155(d). Similarly, proposed

§ 50.34(b)(13) would require applicants for an operating license to submit a final description of the design features included in the plant design in the final safety analysis report under proposed § 50.155(d), and an explanation of how those design features comply with the requirements of proposed § 50.155(d).

Proposed § 50.34(i) would require each application for an operating license to include the applicant's plans for implementing the requirements of proposed § 50.155 and 10 CFR part 50, appendix E, section VII, including a schedule for achieving full compliance with these requirements. This paragraph would also require the application to include a description of: 1) the integrated response capability that would be required by proposed § 50.155(b); 2) the equipment upon which the strategies and guidelines that would be required by proposed § 50.155(b)(1) rely, including the planned locations of the equipment and how the equipment and SSCs would meet the design requirements of proposed § 50.155(c); and 3) the strategies and guidelines that would be required by proposed § 50.155(b)(2).

Proposed § 50.54 Conditions of Licenses

Applicability of the requirements of § 50.54(hh) is currently governed by § 50.54(hh)(3), which makes these requirements inapplicable to a nuclear power plant for which the certifications required under § 50.82(a) or § 52.110(a)(1) have been submitted. This rulemaking proposes to renumber § 50.54(hh)(3) to reflect the proposed movement of the requirements currently within § 50.54(hh)(2) to proposed § 50.155(b)(2). The proposed § 50.54(hh)(2) includes editorial changes to reflect that the applicability is to the licensee rather than the facility and to correct the section numbers for the required certifications. Additionally, proposed § 50.54(hh)(2) clarifies that the inapplicability is dependent upon the NRC docketing of the certifications rather than licensee submittal because § 50.82(a)(2) and § 52.110(b) set the docketing of the certifications as the point at which operation of the reactor is no longer authorized, and fuel cannot be placed in the reactor vessel.

Proposed § 50.155(a), "Applicability"

Proposed § 50.155(a) would describe which entities would be subject to this proposed rule. Proposed § 50.155(a)(1) would provide that each holder of an operating license for a nuclear power reactor under part 50 and each holder of a combined license under part 52 after the Commission has made the finding under § 52.103(g) that the acceptance criteria have been met, would be required to comply with the requirements of this proposed rule until the time when the NRC has docketed the certifications described in § 50.82(a)(1) or § 52.110(a). These certifications inform the NRC that the licensee has permanently ceased to operate the reactor and permanently removed all fuel from the reactor vessel. Upon the docketing of the certifications, by operation of law under § 50.82(a)(2) or § 52.110(b), the licensee's part 50 or 52 license, respectively, no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel. At this point, many portions of this proposed rule would not apply to the licensee because the removal of fuel from the reactor vessel would eliminate the risk of a reactor-based beyond-design-basis event and the need to prepare to mitigate those events. Proposed § 50.155(a)(3) would set forth the requirements that would apply to this licensee.

Proposed § 50.155(a)(2) would provide that each applicant for an operating license for a nuclear power reactor under part 50 and each holder of a combined license before the Commission makes the finding under § 52.103(g) would be required to comply with the requirements of this proposed rule no later than the date on which the Commission issues the operating license under § 50.57 or makes the finding under § 52.103(g), respectively. Under this regulation, operating license applicants and COL holders would be in compliance with this proposed rule before they begin operating their reactors, thereby providing additional defense-in-depth capabilities at the inception of power operations.

Proposed § 50.155(a)(3) would address power reactor licensees that permanently stop operating and defuel their reactors and begin decommissioning the reactors. The proposed paragraph would provide that when an entity subject to the requirements of proposed § 50.155 submits to the NRC the certifications described in section § 50.82(a)(1) or § 52.110(a), and the NRC dockets those certifications, then that licensee would be required to comply with the requirements of proposed § 50.155(b)-(f) associated with maintaining or restoring secondary containment, if applicable, and spent fuel pool cooling capabilities for the reactor described in the § 50.82(a)(1) or § 52.110(a) certifications, except for the requirements in proposed § 50.155(c)(4) and proposed in 10 CFR part 50, appendix E, section VII. In other words, the licensee could discontinue compliance with the requirements in proposed § 50.155 associated with maintaining or restoring core cooling or the primary reactor containment functional capability for the reactor described in the § 50.82(a)(1) or § 52.110(a) certifications. Compliance with the requirements of proposed § 50.155(b)-(f) associated with maintaining or restoring secondary containment, if applicable, and spent fuel pool cooling capabilities would continue as long as spent fuel remains in the spent fuel pool(s) associated with the reactor described in the § 50.82(a)(1) or § 52.110(a) certifications.

Proposed § 50.155(a)(3)(i) would discontinue the requirement to comply with proposed § 50.155(b)(1) requirements concerning beyond-design-basis event strategies and guidelines for spent fuel pool cooling capabilities, and any requirements based on compliance with proposed § 50.155(b)(1), for certain licensees in decommissioning. These licensees would have to perform and retain an analysis demonstrating that sufficient time has passed since the fuel within the spent fuel pool was last irradiated such that the fuel's low decay heat and boil-off period provide sufficient time in an emergency for the licensee to obtain off-site resources to sustain the spent fuel pool cooling function indefinitely and therefore obviate the need to comply with proposed § 50.155(b)(1) using installed or on-site portable equipment.

Proposed § 50.155(a)(3)(i) also would discontinue the requirement to comply with the remaining provisions of proposed § 50.155 except proposed § 50.155(b)(2) when the fuel in the spent fuel pool reaches the point where beyond-design-basis event strategies and guidelines for spent fuel cooling capabilities would no longer be needed.

Proposed § 50.155(a)(3)(ii) would exempt the licensee for Millstone Power Station Unit 1, Dominion Nuclear Connecticut, Inc. from the requirements of proposed § 50.155.

Under proposed § 50.155(a)(3), once a power reactor licensee has permanently stopped operating and defueled its reactor and has removed all irradiated fuel from the spent fuel pool(s) associated with the reactor described in the § 50.82(a)(1) or § 52.110(a) certifications, the licensee could cease compliance with all requirements in proposed § 50.155 for the unit(s) described in the § 50.82(a)(1) or § 52.110(a) certifications.

Proposed paragraph 50.155(a)(4) would specify the persons to whom the requirements of § 50.155(d) apply. Section 50.155(d) would apply to applicants for construction permits, operating licenses, standard design certifications, standard design approvals, combined licenses, and manufacturing licenses. The NRC chose the term “applicants” so that this proposed rule would only apply to a person who has submitted a nuclear power reactor design for NRC approval, but has not yet received NRC approval as of the effective date of this final rule. As a result, this proposed rule would not apply to any licensee, permit holder or applicant for an approved nuclear plant design. For example, under § 50.155(a)(4)(ii), an operating license applicant whose construction permit was issued on or before the effective date of the rule would not be required to comply with § 50.155(d). The Tennessee Valley Authority (TVA) currently holds construction permits for Bellefonte Units 1 and 2. Should TVA submit an application for an operating license for either plant, in neither case would TVA be required to comply with § 50.155(d).

The NRC is also proposing to limit the applicability of § 50.155(d) to only those applications requesting approval of the design of the key safety functions. In doing so, the proposed § 50.155(d) would not apply to an application that references a previously approved nuclear power plant design. For example, under § 50.155(a)(4)(v), a combined license application referencing an existing design certification would not be required to comply. As a result, the South Texas Project Units 3 and 4 combined license application would not be required to comply with § 50.155(d) because it references an existing design certification.

Further, § 50.155(d) would apply to applications for new standard design certifications. However, § 50.155(d) would not apply to applications for renewals of standard design certifications. For example, the NRC has docketed two applications for renewal of the Advanced Boiling Water Reactor design certified in appendix A to 10 CFR part 52. These applicants would not be required to comply with § 50.155(d).

The NRC also clarifies that, while § 50.155(d) would only apply to certain applicants as described in § 50.155(a)(4), future operating and combined licensees would also be required to address the remaining requirements under § 50.155 at the appropriate time as described in § 50.155(a)(2). Further, applicants for an operating license or combined license would also be subject to the information collection requirements in the respective content of applications sections amended in this proposed rule. Therefore, a combined license applicant not referencing a standard design who is subsequently granted a license, and an operating license applicant, would be subject to all of the proposed requirements under § 50.155, whereas other applicants or licensees would only be subject to certain proposed requirements.

Proposed § 50.155(b), "Integrated response capability"

Proposed paragraph (b) would require that each applicant or licensee develop, implement, and maintain an integrated response capability that includes: 1) mitigation strategies for beyond-design-basis external events, 2) severe accident management guidelines, 3) extensive damage mitigation guidelines, 4) integration of these strategies and guidelines with emergency operating procedures, 5) sufficient staffing to support implementation of the guidelines in conjunction with the EOPs, and 6) a supporting organizational structure with defined roles, responsibilities, and authorities for directing and performing these strategies, guidelines, and procedures. The intent is to require that the operating and combined license holders described in § 50.155(a) be able to mitigate the consequences of a wide range of initiating events and plant damage states that can challenge public health and safety.

The specification of strategies, guidelines and procedures for the response capability not only defines the required scope of the capability but sets forth the expectation that the response capability must include planned methods for responding that are documented in some form of written instruction. To serve their function, these strategies, guidelines and procedures must be acted upon by individuals capable of understanding their appropriate application and implementing them. Accordingly, proposed § 50.155(b)(5), in conjunction with proposed § 50.155(e), would require that the response capability include an adequate number of personnel with the knowledge and skills to implement the strategies, guidelines and procedures and that the mitigation activities of these individuals be coordinated in accordance with a defined command and control structure as would be required by proposed § 50.155(b)(6).

Proposed § 50.155(b) would specify that the integrated response capability be "developed, implemented, and maintained." This language reflects NRC consideration that whereas certain elements of the integrated response capability have been developed and are currently in place (e.g., the EOPs), other elements (e.g., guidelines to mitigate beyond-design-basis external events) may require additional efforts to complete and integrate. The term “implement” is used in proposed § 50.155(b) to mean that the integrated response capability is established and available to respond, if needed (e.g., the licensee has approved the strategies, guidelines, and procedures for use). The term “maintain” as used in proposed § 50.155(b) reflects the NRC’s intent that licensees ensure that the integrated response capability, once established, be preserved consistent with the change control provisions of proposed § 50.155(g).

Proposed § 50.155(b)(1) would establish requirements for applicants and licensees to develop, implement and maintain strategies and guidelines to mitigate beyond-design-basis external events from natural phenomenon that result in an extended loss of ac power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. These provisions would require that the strategies and guidelines be capable of being implemented site-wide and include:

i. Maintaining or restoring core cooling, containment, and spent fuel pool cooling capabilities; and

ii. Enabling the use and receipt of offsite assistance and resources to support the continued maintenance of the functional capabilities for core cooling, containment, and spent fuel pool cooling indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies.

New reactors may establish different approaches from operating reactors in developing strategies to mitigate beyond-design-basis events. For example, new reactors may use installed plant equipment for both the initial and long-term response to an ELAP with less reliance on portable equipment and offsite resources than currently operating nuclear power plants. The NRC will consider the specific plant approach when evaluating the SSCs relied on as part of the mitigating strategies for beyond-design-basis events. Additional information on these strategies is provided in DG-1301, which would endorse an updated version of the industry guidance, for use by applicants and licensees, that incorporates lessons learned and feedback stemming from the implementation of Order EA-12-049, consistent with Commission direction.

The proposed § 50.155(b)(1) would limit the requirements for mitigation strategies to addressing "external events from natural phenomenon." This proposed language is meant to differentiate these requirements from those that currently exist within § 50.54(hh)(2), which address beyond-design-basis external events leading to loss of large areas of the plant due to explosions and fire. This proposed provision also results in the need to have mitigation equipment be reasonably protected from the effects of external natural phenomenon as discussed in later portions of this proposed notice.

The proposed requirements to enable "the use and receipt of offsite assistance and resources to support the continued maintenance of the functional capabilities for core cooling, containment, and spent fuel pool cooling indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies" means that licensees would need to plan for obtaining sufficient resources (e.g., fuel for generators and pumps, cooling and makeup water) to continue removing decay heat from the irradiated fuel in the reactor vessel and spent fuel pool as well as to remove heat from containment as necessary until alternate means of removing heat is established. The alternate means of removing heat could be achieved through repairs to existing SSCs, commissioning of new SSCs, or reduction of decay heat levels through the passage of time sufficient to allow heat removal through losses to the ambient environment. More detailed planning for offsite assistance and resources would be necessary for the initial period following the event; less detailed planning would be necessary as the event progresses and the licensee can mobilize additional support for recovery.

Proposed § 50.155(b)(2) would move requirements for EDMGs that currently exist in § 50.54(hh)(2) to proposed § 50.155(b)(2). This move would consolidate the requirements for beyond-design-basis strategies and guidance into a single section to promote efficiency in their consideration and allow for better integration. Although the wording of proposed § 50.155(b)(2) differs from that of § 50.54(hh)(2), no substantive change in the requirements is intended.

The preamble to § 50.155(b)(2) that is contained in § 50.155(b) is worded so that it would require that licensees “develop, implement, and maintain” the strategies and guidance required in § 50.155(b)(2) rather than using the wording of § 50.54(hh)(2) to require that licensees “develop and implement” the described guidance and strategies. The addition of the word “maintain” was proposed in order to correct an inconsistency with the wording of § 50.54(hh)(1), which was promulgated along with § 50.54(hh)(2) in the Power Reactor Security Rulemaking, issued on March 27, 2009 (74 FR 13926), and to clarify that the NRC considers the plain language meaning of the transitive verb “to implement,” “to put into effect,” as it was used in the context of § 50.54(hh)(2) as including maintenance of the resulting guidance and strategies. The wording of the requirement as it was originally issued in the Interim Compensatory Measures Order, EA-02-026, dated February 25, 2002, was worded to require licensees to “develop” specific guidance, while the corresponding license conditions imposed by the conforming license amendment was worded to require each affected licensee to “develop and maintain” strategies. The NRC believes that the phrase “develop, implement, and maintain” would provide better clarity of what is necessary for compliance with the requirements without substantively changing the requirements.

Proposed § 50.155(b)(3) would establish requirements for licensees to develop, implement and maintain strategies and guidelines to mitigate the consequences of events that progress to imminent or actual significant damage to fuel in the reactor vessel or spent fuel pool, to include strategies and guidelines to support actions intended to:

i. Arrest the progression of fuel damage,

ii. Maximize the duration for which containment capability is maintained, and,

iii. Minimize radiological releases.

The phrase “arresting the progression of fuel damage” means stabilizing the condition of fuel damage and preventing further fuel damage. Examples of these actions could be restoring a means of recovering the reactor vessel water level or refilling the spent fuel pool such that sufficient heat can be removed to stabilize conditions and prevent further fuel degradation. The phrase “maximizing the duration for which containment capability is maintained” means removing sufficient mass and energy from containment to prevent the ultimate failure of the containment which could lead to an uncontrolled radiological release to the environment. One example of an action that accomplishes this would be venting the containment only as necessary to maintain containment pressure slightly below the Primary Containment Pressure Limit (i.e., the pressure limit at which containment integrity can be assured assuming maximum decay heat load) thereby maintaining the containment integrity capability as long as possible which also serves to minimize radiological releases to the environment. The phrase “minimizing radiological releases” means that the strategies and guidelines contain guidance to initiate radiological releases in a manner which has the least impact to public health and safety yet still protects against the uncontrolled loss of the containment function. Examples could include direction to vent only as necessary to prevent exceeding the containment design pressure limit or to vent the containment through connections that provide for scrubbing of the vented atmosphere prior to release.

Proposed § 50.155(b)(4) would establish requirements for licensees to integrate the strategies and guidelines in (b)(1) through (b)(3) with EOPs. The Commission’s intent regarding integration of strategies, guidelines, and procedures was introduced in the section-by-section analysis of the proposed § 50.155(b) requirement for an integrated response capability and is described further under "Integration with EOPs" of Section D, Proposed Rule Regulatory Bases.

Proposed § 50.155(b)(5) would establish requirements for licensees to provide the staffing necessary for having an integrated response capability to support implementation of the strategies and guidelines in proposed (b)(1)-(b)(3). The number and composition of the response staff should be sufficient to implement mitigation strategies intended to maintain or restore the functions of core cooling, containment, and spent fuel pool cooling for all affected units. The word "sufficient" is used in the proposed paragraph to reflect its meaning "adequate."

Proposed § 50.155(b)(6) would establish requirements for licensees to have a supporting organizational structure with defined roles, responsibilities, and authorities for directing and performing the guidelines in (b)(1)-(b)(3).

Proposed § 50.155(c) Equipment requirements

Proposed § 50.155(c)(1) would require that equipment relied on for the mitigation strategies of proposed paragraph (b)(1) have sufficient capacity and capability to simultaneously maintain or restore core cooling, containment, and spent fuel pool capabilities for all the power reactor units and spent fuel pools within the licensee's site boundary.

The phrase sufficient "capacity and capability" in proposed § 50.155(c)(1) means that the equipment and the instrumentation relied on to support the decision making necessary to accomplish the associated mitigating strategies of § 50.155(b)(1) should have the design specifications necessary to assure that it will function and provide the requisite plant information when subjected to the conditions it is expected to be exposed to in the course of the execution of those mitigating strategies. These design specifications would include appropriate consideration of environmental conditions that are predicted in the thermal-hydraulic and room heat up analyses used in the development of the mitigating strategies responsive to § 50.155(b)(1).

Proposed § 50.155(c)(2) would require reasonable protection of the § 50.155(b)(1) equipment rather than the treatment of SSCs important to safety under GDC-2, which requires that those SSCs be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. The phrase “reasonable protection” was initially proposed in Recommendation 4.2 of the NTTF Report in the context of a proposed NRC Order to licensees to require “reasonable protection” of equipment required by § 50.54(hh)(2) from the effects of design-basis external events along with providing additional sets of equipment as an interim measure during a subsequent rulemaking on prolonged SBO. The NTTF based this recommendation on the potential usefulness of the EDMGs in circumstances that do not involve loss of a large area of the plant and explained that reasonable protection from external events as used in the NTTF Report meant that the equipment must “be stored in existing locations that are reasonably protected from significant floods and involve robust structures with enhanced protection from seismic and wind-related events.”

The NRC carried forward the use of the phrase “reasonable protection” in Order EA-12-049 with regard to the protection required for equipment associated with the mitigation strategies. That Order did not, however, define “reasonable protection.” The NRC guidance in JLD-ISG-2012-01 discussed “reasonable protection” as follows:

> Storage locations chosen for the equipment must provide protection from external events as necessary to allow the equipment to perform its function without loss of capability. In addition, the licensee must provide a means to bring the equipment to the connection point under those conditions in time to initiate the strategy prior to expiration of the estimated capability to maintain core and spent fuel pool cooling and containment functions in the initial response phase.

In JLD-ISG-2012-01, the NRC endorsed NEI 12-06, Revision 0, as providing an acceptable method to provide reasonable protection, storage, and deployment of the equipment associated with Order EA-12-049. The NEI 12-06, Revision 0, also omitted a definition for the phrase “reasonable protection,” but did provide guidelines for use by licensees for protecting the equipment from the hazards that would be commonly applicable: 1) seismic hazards; 2) flooding hazards; 3) severe storms with high winds; 4) snow, ice and extreme cold; and 5) high temperatures. These guidelines included the use of structures designed to or evaluated equivalent to American Society for Civil Engineers (ASCE) Standard 7-10, “Minimum Design Loads for Buildings and Other Structures,” for the seismic and high winds hazards, rather than requiring the use of a structure that meets the plant’s design basis for the Safe Shutdown Earthquake or high winds hazards including missiles. The NEI 12-06 guidelines also allow storage of the equipment above the flood elevation from the most recent site flood analysis, storage within a structure designed to protect the equipment from the flood, or storage below the flood level if sufficient time will be available and plant procedures would address the need to relocate the equipment above the flood level based on the timing of the limiting flood scenario(s). The NEI 12-06 guidelines further provide that multiple sets of equipment may be stored in diverse locations in order to provide assurance that sufficient equipment will remain deployable to assure the success of the strategies following an initiating event. The NRC-endorsed guidelines in NEI 12-06 do not consider concurrent, unrelated beyond-design-basis external events to be within the scope of the initiating events for the mitigating strategies. There is an assumption of a beyond-design-basis external event that establishes the event conditions for reasonable protection, and then it is assumed that the event leads to an ELAP and LUHS. But, for example, there is not an assumption of multiple beyond-design-basis external events occurring at the same time. As a result, reasonable protection for the purposes of compliance with Order EA-12-049 would allow the provision of specific sets of equipment for specific hazards with the required protection for those sets of equipment being against the hazard for which the equipment is intended to be used.

The NRC proposes to continue the use of the phrase “reasonable protection” in proposed § 50.155(c)(2) in order to distinguish the character of the required protection of GDC-2, which requires that SSCs important to safety be designed to withstand the effects of natural phenomena, from that of proposed § 50.155(c)(2), which would allow damage to or loss of specific pieces of equipment so long as the capability to use some of the equipment to accomplish its intended purpose is retained. “Reasonable protection” would also allow for protection of the equipment using structures that could deform as a result of natural phenomena so long as the equipment could be deployed from the structure to its place of use.

The remaining portion of proposed § 50.155(c)(2) would set the hazard level for which “reasonable protection” of the equipment must be provided. The hazard level would be the level determined for the design basis for the facility for protection of safety-related SSCs from the effects of natural phenomena, or, for the seismic or flooding hazards, the greater of the hazard level determined for the design basis for the facility and the licensee’s reevaluated hazards, stemming from the March 12, 2012, NRC letter issued under § 50.54(f). The timing for the proposed requirement for reasonable protection against the reevaluated hazards is set by § 50.155(c)(2)(i) at two years following the effective date of the proposed rule. Operating power reactor licensees that were requested to reevaluate their seismic and flooding hazard levels by the NRC by letter dated March 12, 2012 under 10 CFR 50.54(f) are currently on a submittal and NRC review schedule to have confirmation of the reevaluated hazard levels by December 2015. Given that the rulemaking schedule for this proposed rule is to provide the final rule to the Commission in December 2016, the anticipated effective date of the final rule would be mid-to-late 2017. Requiring compliance within two years following the effective date of the final rule would allow licensees with a new hazard level the opportunity to take measurements to support any necessary plant modifications during the first refueling outage following NRC confirmation of those levels and the opportunity to implement those modifications in a subsequent refueling outage after the effective date of the rule. The NRC is requesting feedback on this proposed implementation schedule in section VII of this notice.

Proposed paragraph (c)(3) would require that licensees perform adequate maintenance on the equipment relied on for the mitigation strategies responsive to proposed paragraph (b)(1) to assure that the equipment is capable of fulfilling its intended function following a beyond-design-basis external event. The phrase “adequate maintenance” means sufficient routine maintenance and testing are performed, reflecting the storage and readiness conditions of the equipment, for a licensee to conclude that the equipment is capable of performing its function to a degree that would support the successful execution of the mitigation strategies of paragraph (b)(1). Provision of “adequate maintenance” also entails the establishment of a system of programmatic controls for the equipment to limit the quantity of equipment taken out of service for maintenance and testing in order to limit the unavailability of that equipment appropriately and to provide assurance that sufficient equipment will remain available to satisfy proposed paragraph (c)(1).

Proposed paragraph (c)(4) would make generically applicable the requirements of Order EA-12-051 by requiring that licensees include a reliable means to remotely monitor wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions.

Proposed § 50.155(d) Design features

Proposed § 50.155(d), would require that each applicant specified in § 50.155(a)(4) shall include design features in the plant design sufficient to enhance coping durations and minimize reliance on human actions to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities during an ELAP concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. The proposed requirement would require an applicant for a future license, certification, or approval for a nuclear power plant design to consider the effects of an ELAP early in the design process. This paragraph would require new reactor designs to include design features that provide enhanced capabilities for addressing these events. The term “design features” means SSCs, including the physical arrangement of such SSCs, and their functional capabilities—key characteristics of the SSCs that result in their contribution to maintain or restore the key safety functions. Examples of design features are the inclusion of equipment such as passive or steam-driven cooling systems that might be supported by dc power. Examples of functional capabilities of a design feature are the flow capacity of a pump or the electrical capacity of a power supply.

Proposed § 50.155(d) would also require that the applicant include design features in the plant design sufficient to address the effects of the events. The term “include design features in the plant design” means that the plant design would include those design features that support the key safety functions. For example, if the designer determines that the key safety functions would be more effectively maintained by including a specific installed component rather than relying on portable equipment, the plant design could include a design feature.

Further, § 50.155(d) would require design features that enhance coping durations and minimize reliance on human actions to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities. The phrase “enhance coping durations” in proposed § 50.155(d) means that the design features increase the amount of time that core cooling, containment, and spent fuel pool cooling capabilities can be maintained in the initial phase of the event before the transition from the permanently installed SSCs to portable equipment. The term “minimize reliance on human actions” means that the design features either obviate or reduce, to the extent practical, the need for operator actions during the initial phase of the event. The NRC chose “minimize” versus “reduce” in paragraph (d) to require the applicant to design the plant to cope with the ELAP condition with as little human action as is practical (i.e., minimize) rather than less human action (i.e., reduce). The NRC intends for the designer to minimize human actions during this condition so that: 1) operators would be able to focus more time and attention on monitoring plant conditions and planning for the potential transition to relying on portable equipment and offsite resources; and 2) there is lower potential for human failures during adverse, stressful conditions with the objective of achieving a lower risk of failure to execute the mitigation strategy.

The condition described in § 50.155(d) is an ELAP concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. This condition is consistent with that of § 50.155(b)(1) as it applies to mitigation strategies for holders of operating licenses and combined licenses. Refer to the section by section analysis for § 50.155(b)(1) for a discussion of the condition. The rule language in this paragraph addresses designs that rely on either the ultimate or the normal heat sink.

Proposed § 50.155(e) Training requirements

Proposed § 50.155(e) would require that each licensee specified in § 50.155(a) provide for the training and qualification of personnel that perform activities in accordance with the strategies and guidelines identified in § 50.155(b)(1)-(b)(3).

Proposed § 50.155(f) Drills and Exercises

Proposed § 50.155(f) would require that each licensee and applicant specified in § 50.155(a) conduct drills and exercises for personnel, who would perform activities in accordance with the strategies and guidelines identified in § 50.155(b)(1)-(b)(3). The use of drills and exercises allows demonstration and evaluation of the licensee's capability to execute the integrated response capability required by § 50.155(b) mitigation strategies and guidelines in light of the specific plant damage and operational conditions presented by an initiating event. "Integrated" is used to describe the licensee's or applicant's approach to using all tools, spaces, qualified personnel and resources during a performance enhancing experience to the furthest extent practical given a set of initiating conditions and within the bounds of a drill or exercise scenario. When two or more strategies or guidelines in § 50.155(b)(1)-(b)(3) are potentially useful, "integrated" is meant that transitions to and from one set of strategies or guidelines in § 50.155(b)(1)-(b)(3) to another are coordinated.

This proposed rule uses the words "drill" and "exercise" as they are defined in NUREG-0654/FEMA-REP-1, Revision 1,[8] meaning an evaluated performance-enhancing experience that reasonably simulates the interactions between appropriate centers, work groups, strike teams, or individuals that would be expected to occur during the event. For the initial drill or exercise, the licensee would be required to demonstrate its capability to transition to and use one or more of the strategies that would be required by § 50.155(b)(1)-(b)(3) from the AOPs or EOPs, whichever would govern for the initiating event and plant degrading conditions, using the equipment and communication systems used for the EOPs and guidelines.

Proposed § 50.155(f)(1) would require the initial drill or exercise to be conducted within 12 months prior to the issuance of the first operating license (OL) for the unit described in the application. This would allow the license applicant to implement any improvements or corrective actions identified during the drill or exercise, and allow the Commission to consider the results of any drill or exercise actions in the decision whether to authorize the OL. Because § 50.155(f)(1) applies only to applicants for operating licenses, it would not apply to holders of

[8] Planning Standards N.1 Exercise and N.2 Drills.

operating licenses under 10 CFR part 50, who are subject to proposed § 50.155(f)(4), or holders of combined licenses under 10 CFR part 52, who are subject to proposed § 50.155(f)(2) - (4).

Proposed § 50.155(f)(2) would require the licensee to conduct an initial drill or exercise that demonstrates the capability to transition from the AOPs or EOPs, use one or more of the strategies and guidelines in paragraphs (b)(1)-(b)(3) of this section, and use communications equipment required in 10 CFR part 50, appendix E, section VII no more than 12 months before the date specified for completion of the last inspections, tests, and analyses in the inspections, tests, analyses, and acceptance criteria (ITAAC) completion schedule as required by § 52.99(a) for the unit described in the combined license.

This proposed rule would set the completion date for the initial drill or exercise at "no more than 12 months before the date specified for completion of the last inspections, tests, and analyses in the ITAAC completion schedule required by § 52.99(a) for the unit described in the combined license" in order to allow the licensee to implement any improvements or corrective actions identified during the drill or exercise, and allow the Commission to consider the results of any drill or exercise actions in the decision whether to make the § 52.103(g) finding.

The proposed § 50.155(f)(2) requirement for initial drills or exercises is limited to holders of combined licenses under 10 CFR part 52 before the Commission has made the finding under § 52.103(g). A combined license holder for whom the Commission has already made the finding under § 52.103(g) as of the effective date of the rule would not be subject to proposed § 50.155(f)(2), but would instead be subject to § 50.155(f)(4) for the proposed initial drill requirements. Proposed § 50.155(f)(3) would require operating power reactor licensees under 10 CFR part 50 and those under 10 CFR part 52 for whom the Commission has made the finding under § 52.103(g) to conduct subsequent drills, exercises, or both that collectively demonstrate a capability to use at least one of the strategies and guidelines in each of proposed § 50.155(b)(1)-(b)(3) in succeeding 8-year intervals. This would require that the drills and exercises performed to demonstrate this capability include transitions from other procedures and guidelines as applicable, and the use of communications equipment that would be required by proposed 10 CFR part 50, appendix E, section VII. This proposed requirement differs from the proposed § 50.155(f)(1) and (f)(2) initial demonstration requirement, in that it would require licensees to demonstrate a continuing capability, and as such, it is structured to require licensees to demonstrate at least one of the strategies and guidelines from each of the guidelines during the 8-year interval.

Proposed § 50.155(f)(4) would require holders of operating licenses or combined licenses for which the Commission has made the finding under § 52.103(g) to conduct an initial drill or exercise that demonstrates the capability to transition to and use one or more of the strategies and guidelines in proposed § 50.155(b)(1)-(b)(3) and use communications equipment required in 10 CFR part 50, appendix E, section VII. Proposed § 50.155(f)(4) would be equivalent to proposed § 50.155(f)(1) and (2) for initial drills or exercises, but would apply to current licensees. Following this initial drill or exercise, the licensee would be required to conduct subsequent drills, exercises, or both that collectively demonstrate a capability to use at least one of the strategies and guidelines in each of proposed § 50.155(b)(1)-(b)(3) in succeeding 8-year intervals. Proposed § 50.155(f)(4) would be equivalent to proposed § 50.155(f)(3) for subsequent drills or exercises, but would apply to current licensees under 10 CFR part 50 and those under 10 CFR part 52 for whom the Commission has made the finding under § 52.103(g) as of the effective date of the rule.

Proposed § 50.155(g) Change Control

Proposed § 50.155(g) would establish requirements that govern changes in the implementation of the requirements of proposed § 50.155 and 10 CFR part 50, appendix E, section VII. Prior to implementing a proposed change, proposed § 50.155(g)(1) would require the licensee to perform an evaluation to ensure that the provisions of proposed § 50.155 and 10 CFR part 50, appendix E, section VII, continue to be met. Proposed § 50.155(g)(2) would require that licensees maintain documentation of the paragraph (g)(1) evaluations until the requirements of this proposed § 50.155 and 10 CFR part 50, appendix E, section VII, no longer apply. Finally, proposed § 50.155(g)(3) would inform licensees that proposed changes must continue to be subject to all other applicable change control processes.

Proposed § 50.155(g)(4) would require a licensee who was, as an applicant, subject to § 50.155(d), to ensure that any changes to the design features included in the plant design under § 50.155(d) continue to satisfy the design objectives described in § 50.155(d).

Proposed § 50.155(h) Implementation

Proposed § 50.155(h) would set schedules for compliance for different classes of licensees depending on the circumstances unique to each class. Paragraphs (h)(1) and (2) would require licensees of operating reactors to comply with all requirements within four years of the effective date of the rule.

Proposed 10 CFR Part 50, Appendix E, Section I, Introduction

The NRC proposes adding the sentence, “Section VII of this appendix also provides for ‘Communications and Staffing Requirements for the Mitigation of Beyond-Design-Basis Events’ that do not need to be contained within a licensee’s emergency plan” to the end of paragraph I.2. The NRC is not proposing to require an applicant or licensee to address or implement the proposed requirements in Section VII of Appendix E through the applicant’s or licensee’s emergency plan or maintain the capabilities as a part of the emergency preparedness program. This would allow for site-specific flexibility in implementation.

Proposed 10 CFR Part 50, Appendix E, Section IV.B, Assessment Actions

The NRC proposes adding the phrase, "including from all reactor core and spent fuel pool sources," into paragraph B.1 following "determining the magnitude of, and for continually assessing the impact of, the releases of radioactive materials." This proposed rule would require all licensees to establish the capability to perform offsite dose assessments during an event involving concurrent radiological releases from all on-site units and spent fuel pools, and for multiple release points. The capability would quantify the total releases from the site and estimate the offsite dose consequences.

Proposed 10 CFR Part 50, Appendix E, Section IV.E, Emergency Facilities and Equipment

The NRC proposes adding the phrase, "including from all reactor core and spent fuel pool sources," into paragraph E.2 following "equipment for determining the magnitude of, and for continuously assessing the impact of, the release of radioactive materials to the environment." The proposed rule would require that equipment used for multi-unit dose assessment be maintained in a ready state.

Proposed 10 CFR Part 50, Appendix E, Section IV, Training

The proposed rule would move the § 50.54(hh)(2) exercise requirement from 10 CFR part 50, appendix E, section IV.F.2.j to § 50.155(f). This move would change the exercise requirement to a drill requirement, aligning the requirement with the mitigation strategies drill requirements described in § 50.155(f).

The proposed rule would also require that periodic opportunities for a performance enhancing experience should be provided to personnel responsible for performing multi-unit dose assessment and assessing the results in accordance with the site's emergency plan and implementing procedures.

Proposed 10 CFR Part 50, Appendix E, Section VI, Emergency Response Data Systems

The NRC proposes to change its Emergency Response Data Systems regulations to require the use of technology-neutral equipment. The NRC proposes to restate the requirements in paragraph 3.c to replace the phrase “onsite modem” with “equipment” and removing references to a specific “unit” or equipment use.

Proposed 10 CFR Part 50, Appendix E, Section VII, Communications and Staffing Requirements for the Mitigation of Beyond-Design-Basis Events

Proposed section VII would require power reactor applicants and licensees to conduct a detailed analysis to provide the basis for the staffing necessary for responding to a beyond-design-basis external event as described in § 50.155(b)(1) during an extended loss of ac power (ELAP), and while access to the plant and normal access to the ultimate or normal heat sink are lost. Additionally, the proposed section VII would require power reactor applicants and licensees to maintain at least one onsite and one offsite communications system functional during an ELAP and a loss of the local communication infrastructure.

The current rule in 10 CFR part 50, appendix E, section IV.E.9 requires, “At least one onsite and one offsite communication system; each system shall have a backup power source.” However, the current rule doesn’t address an interruption in the offsite communication services. The proposed rule would require the power reactor applicants and licensees to maintain the communication capabilities of communication amongst onsite staff and between onsite staff and offsite personnel in light of the lessons learned at Fukushima Dai-ichi. Furthermore, the proposed rule would require the power reactor applicants and licensees to submit the staffing analysis, results and implementation plans to meet the requirements, and the submissions would afford the NRC the opportunity to identify any common industry implementation problems and address them in guidance.

The proposed rule would require an applicant for an OL to complete a detailed staffing analysis at least 2 years before the issuance of the first operating license for full power (one authorizing operation above 5 percent of rated thermal power). The time frame allows the applicant to implement any improvements or corrective actions identified during the analysis, and the results of any analysis to inform the Commission's decision in authorizing the OL.

The proposed rule would require that an applicant for a COL conduct a detailed staffing analysis and submit the analysis and results to the NRC 2 years before the date specified for completion of the last inspections, tests, and analyses in the inspections, tests, analyses, and acceptance criteria (ITAAC) completion schedule as required by § 52.99(a) for the unit described in the combined license. The time frame allows the applicant to implement any staffing and communications system improvements and corrective actions identified during the analysis, and the results of any analysis to inform the Commission's decision in issuing the § 52.103(g) finding.

The proposed rule would provide that when the NRC has docketed the certifications described in § 50.82(a)(1) or § 52.110(a) for a power reactor licensee, then that licensee would no longer be subject to section VII of appendix E to 10 CFR part 50 for the unit described in the § 50.82(a)(1) or § 52.110(a) certifications.

Proposed § 52.47 Contents of Applications; Technical Information

Section 52.47 identifies the required technical information to be included in an application for a standard design certification. The proposed rule would revise this section by adding a new paragraph (a)(29) requiring each application for a standard design certification subject to proposed § 50.155(d) to include, in the FSAR, a description of the design features included in the plant design under proposed § 50.155(d) and an explanation of how those design features would comply with the requirements of proposed § 50.155(d). This requirement would apply only to standard design certification applications, and would not apply to standard design certifications issued before the effective date of this rule. Thus, any standard design certification application that is docketed and is under review by the NRC as of the effective date of this proposed rule would need to amend its application to include the information required by proposed § 50.155(d).

Proposed § 52.79 Contents of Applications; Technical Information in Final Safety Analysis Report

Section 52.79 identifies the required technical information to be included in an application for a combined license. The proposed rule would revise this section by adding a new paragraph (a)(48) requiring each application for a combined license that is subject to proposed § 50.155(d) to include a description of the design features included in the plant design under proposed § 50.155(d) and an explanation of how those design features would comply with the requirements of proposed § 50.155(d). The proposed requirement would not apply to combined licenses issued before the effective date of this rule, nor would it apply to combined licenses referencing a standard design certification, a standard design approval, or a manufacturing license. Thus, a combined license application, not referencing either a standard design certification, a standard design approval, or a manufacturing license, that is docketed and under review by the NRC as of the effective date of this proposed rule would need to be amended to include the information required by proposed § 50.155(d).

Proposed § 52.80 Contents of Applications; Additional Technical Information

Section 52.80 identifies the required additional technical information to be included in an application for a combined license. Proposed paragraph (d) would be amended to require a combined license applicant to include the applicant's plans for implementing the requirements of proposed § 50.155 and 10 CFR part 50, appendix E, section VII, including a schedule for achieving full compliance with these requirements. This paragraph would also require the application to include a description of: 1) the integrated response capability that would be required by proposed § 50.155(b); 2) the equipment upon which the strategies and guidelines that would be required by proposed § 50.155(b)(1) rely, including the planned locations of the equipment and how the equipment and SSCs would meet the design requirements of proposed § 50.155(c); and 3) the strategies and guidelines that would be required by proposed § 50.155(b)(2).

Proposed § 52.137 Contents of Applications; Technical Information

Section 52.137 identifies the required technical information to be included in an application for a standard design approval. The proposed rule would revise this section by adding a new paragraph (a)(27) that would require each application for a standard design approval subject to proposed § 50.155(d) to include in the FSAR a description of the design features included in the plant design under proposed § 50.155(d) and an explanation of how those design features would comply with the requirements of proposed § 50.155(d). This requirement would apply only to standard design approval applications, and would not apply to standard design approvals issued before the effective date of this rule. Thus, any standard design approval application that is docketed and under review by the NRC as of the effective date of this rule would need to be amended to include the information required by proposed § 50.155(d).

Section § 52.157 Contents of Applications; Technical Information in the Final Safety Analysis Report

Section 52.157 identifies the required technical information to be included in the final safety analysis report as part of an application for a manufacturing license. The proposed rule would revise this section by adding a new paragraph (f)(33) that would require each application for a manufacturing license that is subject to proposed § 50.155(d) to include a description of the design features included in the plant design under proposed § 50.155(d) and an explanation of how those design features would comply with the requirements of proposed § 50.155(d). Thus, the proposed requirement would not apply to manufacturing licenses issued before the effective date of this rule, nor would it apply to manufacturing license applications referencing a standard design certification or standard design approval. Thus, any manufacturing license application, not referencing a standard design, that is docketed and under review by the NRC as of the effective date of this proposed rule would need to be amended to include the information required by proposed § 50.155(d).

VI. Regulatory Oversight of Severe Accident Management Guidelines.

As discussed in section IV.B. of this notice, the NRC's intent for regulatory oversight of the SAMGs is limited to oversight through inspection. The NRC considered including the SAMGs as an inspection program element of the baseline inspection program of the reactor oversight process (ROP) under the emergency preparedness cornerstone during the development of the ROP. At that time, the scope of the inspection program element considered was documented as covering the use of strategies for dealing with accidents that impact reactor coolant system integrity, with the scope considered as verifying that the SAMGs are written in such a manner as to not impede implementation of the emergency plan. Inspection Manual

Chapter (IMC) 0308, “Reactor Oversight Process Basis Document,” Attachment 2, “Technical Basis for Inspection Program,” documents that the NRC “staff concluded that regular inspection of SAMG was not appropriate because the guidelines are voluntary and have no regulatory basis. The emergency response organization that would implement SAMGs is inspected through EP [emergency preparedness] baseline inspection and performance is covered by two PIs [performance indicators].”

Subsequent to the Fukushima Dai-ichi accident, the NRC performed an inspection under TI-2515/184, “Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs),” in order to determine that the SAMGs were available and how they were being maintained and to determine the nature and extent of licensee implementation of SAMG training and exercises. The results of the inspections conducted under TI-2515/184 informed the NTTF recommendations and the NRC approach for the proposed regulatory oversight of SAMGs. This proposed rule would require licensees to develop, implement, and maintain SAMGs, and as such, the NRC would now include SAMGs within the scope of the ROP.

The NRC’s intent for inspection of SAMGs is similar in nature to the inspections that were conducted under TI-2515/184 and those that are conducted as part of the baseline inspection program for the mitigating strategies under the current § 50.54(hh)(2) using Inspection Procedure (IP) 71111.05T, “Fire Protection (Triennial),” ADAMS Accession No. ML12328A158, and IP 71111.05XT, “Fire Protection - NFPA 805 (Triennial),” ADAMS Accession No. ML12328A167. The NRC will use the results of the inspections performed under TI-2515/191, “Inspection of the Licensee's Responses to Mitigation Strategies Order EA-12-049, Spent Fuel Pool Instrumentation Order EA-12-051 and Emergency Preparedness Information Requested in NRC March 12, 2012,” to inform the development of the inspection procedure for SAMGs and will ultimately use IMC 0308, Appendix B, “Reactor Oversight Process

Realignment," ADAMS Accession No. ML112990461, to determine the appropriate allocation of inspection resources to the resulting inspection procedure.

VII. Specific Requests for Comments.

The NRC is seeking advice and recommendations from the public on the proposed rule. We are particularly interested in comments and supporting rationale from the public on the following:

1. Change Control. The provisions governing change control in proposed § 50.155(g) do not contain a criterion or a set of criteria that would establish a threshold beyond which prior NRC review and approval would be necessary to support a proposed change to the facility impacting the beyond-design-basis aspects of this proposed rulemaking and its supporting implementation guidance. The NRC concluded that a set of threshold criteria that could be used for judging changes to the facility impacting the proposed (b)(1) mitigation strategies would be different from change control criteria governing changes to SAMGs of proposed (b)(3), unless the threshold is established at a higher level. For example, a set of criteria that asks whether a proposed facility change adversely impacts the capability to maintain and restore core cooling, containment, and spent fuel pool cooling capabilities, in conjunction with a criterion that asks whether the proposed facility change adversely impacts the supporting equipment requirements in proposed paragraph (c) might be sufficient for judging whether changes to the facility that impact the implementation of the mitigation strategies of proposed (b)(1) require prior NRC review and approval. However, this set of criteria may not be optimal for SAMGs, and instead a better set of criteria might focus on whether the proposed change adversely impacts the SAMGs from the perspective of supporting informed decision-making in a post-core damage situation. What are stakeholders' views on this proposed change

control structure, and what do stakeholders suggest for revising the change control process to contain criteria for determining the need for prior NRC review and approval?

2. Application of Other Change Control Processes. Proposed § 50.155(g)(3) contains a requirement for licensees to use all applicable change control processes for facility changes, and not simply apply proposed paragraph (g) (i.e., the proposed change control process of paragraph (g) is only applicable to facility changes with respect to their beyond-design-basis aspects and to the extent that such changes impact implementation of the requirements of proposed § 50.155 or the proposed 10 CFR part 50, appendix E, section VII) to the exclusion of other change control processes. This recognizes that facility changes can impact multiple aspects of the plant having different applicable requirements, and being subject to different change control requirements. For example, a licensee may want to make a facility change (e.g., a physical connection device) to support implementation of the beyond-design-basis external event mitigation strategies, and this change might impact safety-related SSCs. In addition to applying the new change control provision to ensure beyond-design-basis aspects of the proposed change result in continued compliance with the new requirements of this proposed rule, the licensee would also need to apply 10 CFR 50.59 to ensure that the facility change does not, due to its impact on safety-related SSCs, require prior NRC approval. The NRC requests feedback on the need for this proposed provision, or suggestions on how it might be improved.

3. Reasonable Protection. The proposed rule language contains a requirement in proposed § 50.155(c)(2) that equipment supporting the proposed mitigation requirements of paragraph (b)(1) be “reasonably protected” from the effects of natural phenomenon including both those in the current plant design basis as well as the

reevaluated hazards under the March 12, 2012, § 50.54(f) request concerning flooding and seismic hazards. As a practical matter, implementation of Order EA-12-049 began before the reevaluated hazard information was available. The NRC recognizes that licensees were mindful of the hazard information, and attempted to address it during implementation. The NRC requests feedback concerning any costs and impacts that licensees would expect to occur as a result of this proposed requirement to include such things as rework or changes to previously implemented mitigation strategies.

4. SAMGs. The NRC is proposing to include SAMGs as requirements in this rulemaking, but recognizes that quantifiable risk information alone is not supportive of the SAMG requirements. Accordingly, to support a fully informed decision for the final rule, the NRC would like stakeholder views on alternative regulatory approaches that can achieve the objective (e.g., ensuring through inspection that SAMGs are updated to reflect the recent post-Fukushima work and are maintained within configuration management programs, or oversight through observation of SAMG drills) while minimizing the regulatory impact. The NRC suggests that input could include alternative regulatory approaches, extended implementation periods, regulatory structures that minimize distraction of resources from more important tasks (for safety), and any cost/impact information that would enable the NRC to update the supporting regulatory analysis.

5. New Reactor Requirements. The proposed rule language includes a requirement for applicants for new nuclear power reactors to include design features in the plant design sufficient to enhance coping durations and minimize reliance on human actions to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities (key safety functions) during an ELAP concurrent with loss of normal access to the normal or ultimate heat sink. The underlying purpose of such a requirement would be

for a new reactor design to have greater reliance on installed equipment, less reliance on operator action, and thus more time for diagnosis, planning, and preparation for the transition to reliance on portable equipment and offsite services. While this requirement would focus on potential improvements in the plant design, it would not obviate the need for a licensee using this design to develop mitigating strategies to depend on portable equipment and offsite resources. Further, this requirement would only apply to applicants who are seeking approval of the key safety functions, and would not apply to an applicant referencing a design where those key safety functions have been previously approved by the NRC. What are stakeholders' views on the need for such a requirement? Is there a better way of accomplishing the aforementioned underlying purpose? Should the requirement be imposed on other applicants or licensees (i.e., regardless of whether the NRC had previously approved the design of the key safety functions), and if so, under what basis? What are stakeholders' views on the costs and benefits of such a requirement? Please provide any available information to support your comments.

6. Transition to SAMGs. Proposed § 50.155(b)(3) would require strategies and guidelines for mitigating the consequences of events that "progress to imminent or actual significant damage" to fuel in the reactor vessel or spent fuel pool. These types of strategies and guidelines have historically been referred to as SAMGs. The NRC understands that in practice, transition to the SAMGs would occur if there are available instrumentation readings (e.g., core exit temperatures exceed 1200 degrees Fahrenheit) or there is an absence of available instrumentation readings but circumstances indicate to the operators that actual or imminent damage to fuel, may have occurred. The actual condition of the fuel is not known, and as such it is not possible to know if "significant

damage" has occurred. The intent of the proposed rule language is to align with practical implementation for transitioning to SAMGs. The NRC requests feedback on whether this proposed language would present problems for licensee implementation or whether it might have other limitations or unintended consequences associated with it. The NRC requests suggestions for improving the language to have it more closely align with actual implementation.

7. Mitigation of Beyond-Design-Basis Events Staffing Analysis. Proposed 10 CFR part 50, appendix E, section VII, would require an analysis for the staffing necessary to support mitigation of a beyond-design-basis external event. This requirement would supplement the separate staffing analysis requirement that already exists in 10 CFR part 50, appendix E, section IV.A.9. The reason for the two separate staffing analysis requirements is related to the historical imposition of the requirements for the staffing analyses in the emergency preparedness rulemaking of 2011 and the March 12, 2012 Request for Information under 10 CFR 50.54(f). The NRC is seeking feedback on whether it would be more efficient in practice for the two staffing analyses and their corresponding requirements to be combined, particularly for future reactor applicants. Would there be any unintended consequences to keeping the analyses separate or combining them? Is there a better way of achieving the underlying purpose of this requirement?

8. Training Requirements. Section 50.155(e) of the proposed rule would require licensees to provide for the training and qualification of personnel that perform activities in accordance with the strategies and guidelines identified in paragraphs (b)(1)–(b)(3) (i.e., mitigation strategies for beyond-design-basis external events, extensive damage mitigation guidelines, and severe accident management guidelines) using the SAT

process as defined in § 55.4. The NRC notes that whereas many individuals at licensee facilities that would be subject to the proposed rule are trained under the SAT process (e.g., individuals specified under § 50.120), some individuals (e.g., firefighting and emergency preparedness personnel) may be currently trained under programs that are not required by NRC regulation to use the SAT process (e.g., National Fire Protection Association standards for training and 10 CFR part 50, appendix E). It is not the NRC's intent to extend the requirement for SAT-based training to the entirety of such programs. Rather, the intent of the proposed requirement would be to ensure that any training that is not currently part of existing programs but would be needed for performing activities in accordance with the strategies and guidelines identified in paragraphs proposed § 50.155(b)(1)–(b)(3) be identified and provided for in accordance with the SAT process. The NRC requests comment on potential unintended consequences of the proposed rule language for programs not currently required to be SAT-based and if unintended consequences are identified, proposed alternative language for requiring the necessary amendments to such programs.

9. Drill or Exercise Frequency. Proposed § 50.155(f)(3) and § 50.155(f)(4) would require that following an initial drill or exercise, licensees would be required to conduct subsequent drills, exercises, or both, that collectively demonstrate a capability to use at least one of the strategies and guidelines in each of proposed § 50.155(b)(1)-(b)(3) in succeeding 8-year intervals. This would require that the drills or exercises performed to demonstrate this capability include transitions from other procedures and guidelines as applicable, and the use of communications equipment that would be required by proposed 10 CFR part 50, appendix E, section VII, and that licensees shall not exceed 8 years between any consecutive drills or exercises. These requirements would be

separate from the 8-year emergency preparedness exercise cycle requirements in 10 CFR part 50, appendix E, section IV.F. The NRC is seeking feedback on whether the drill or exercise frequency proposed by § 50.155(f)(3) and § 50.155(f)(4) is appropriate.

10. Equipment Requirements. Proposed § 50.155(c)(1) would require the capacity and capability of the equipment relied on for the mitigation strategies required by proposed § 50.155 (b)(1) to be sufficient to simultaneously maintain or restore core cooling, containment, and spent fuel pool cooling capabilities for all the power reactor units within the site boundary. Additionally proposed § 50.155(c)(3) would require the equipment relied on for the mitigation strategies in proposed § 50.155(b)(1) to receive adequate maintenance such that the equipment is capable of fulfilling its intended function. The intent of these two proposed provisions is to make elements of Order EA-12-049 generically-applicable. Order EA-12-049 did not contain a specific maintenance requirement, but instead contained a performance-based requirement "to develop, implement and maintain strategies," and failure to perform adequate maintenance would likely lead to a failure to meet this more general requirement, which is also contained in proposed § 50.155(b)(1). Additionally, the supporting guidance for the proposed rule for proposed § 50.155(b)(1) carries forward the same approach that was used for implementation of Order EA-12-049, and contains a number of programmatic controls that in an analogous fashion to the maintenance provision in proposed § 50.155(c)(3), if not followed, would likely lead to a loss of equipment capacity and capability and result in a failure to comply with the proposed § 50.155(b)(1). Therefore, the NRC would like stakeholder views on the need for a separate maintenance provision within the proposed rule.

11. Equipment Protection Implementation Deadline. The NRC is proposing to require licensees to reasonably protect the equipment relied upon to implement the mitigation strategies required by proposed § 50.155(b)(1). That equipment would need to be reasonably protected from the effects of natural phenomena that are, at a minimum, equivalent to the design basis of the facility. The proposed rule would require each licensee that received the March 12, 2012, NRC letter issued under § 50.54(f) to provide reasonable protection against that reevaluated seismic or flooding hazard(s) by 730 days following the effective date of the final rule, if the reevaluated hazard exceeds the design basis of its facility. This is based on the anticipated completion dates for the licensees' hazard reevaluations and their confirmation by the NRC and the potential need for planning and implementing modifications during refueling outages. The NRC recognizes that certain licensees may need input into their analyses of reevaluated hazards from other government agencies, without any certainty of when that input would be provided. This reliance on information from other entities could remove from the licensee's control the ability to comply with the proposed rule by a specific date. The NRC requests comments on the proposed implementation schedule, including suggestions for the criteria that licensees would need to satisfy to extend the schedule.

12. Methodology for addressing reevaluated hazards. In SRM-COMSECY-14-0037, the Commission affirmed that: 1) licensees for operating nuclear power plants need to address the reevaluated flooding hazards within their mitigating strategies for beyond-design-basis external events; and 2) licensees for operating nuclear power plants may need to address some specific flooding scenarios that could significantly damage the power plant site by developing targeted or scenario-specific mitigating strategies, possibly including unconventional measures, to prevent fuel damage in reactor cores or

spent fuel pools. The NRC is proposing to require licensees for operating nuclear power plants to address the reevaluated flooding hazard levels by reasonably protecting the mitigating strategies equipment to those levels if they exceed the design-basis flood level for the facility. Alternatively, the NRC could: 1) place this requirement within § 50.155(b)(1) as a condition the associated strategies and guidelines must be capable of addressing; or 2) include a separate requirement for targeted or scenario-specific mitigating strategies as an option to address the reevaluated flooding hazards. The NRC seeks comment on whether the first of these options would be a better means to communicate the need for a licensee's strategies and guidelines to be capable of execution in the context of the new flooding hazard levels than including the requirement in § 50.155(c)(2). The NRC seeks additional comment on whether it would be appropriate to allow further flexibility in the licensee's strategies and guidelines by establishing an alternative means of compliance that does not include the surrogate condition of a loss of all alternating current power for specific beyond-design-basis conditions such as the reevaluated flooding hazards. For example, if a licensee could protect their internal power distribution system and emergency diesel generators from the reevaluated flooding hazard, it may not be necessary for the licensee to assume the loss of all alternating current power.

13. Command and Control. Requirements for command and control and organizational structures currently exist in numerous locations, including 10 CFR part 50, appendix E, section IV.A as well as within the typical administrative controls portions of technical specifications for power reactor licensees. These requirements do not plainly limit the scope of the roles, responsibilities and authorities to events within the design or licensing basis of the facility, although past NRC practice has been to treat these requirements in that manner. The proposed rule includes a further requirement on the subject in order to

clarify the scope of what is required for organizational structures at power reactor licensees. Alternatively, the NRC is considering if the expansion of scope of regulatory oversight of the organizational structures would require imposition of a new requirement or the expansion of scope would be better accomplished by communicating the understanding that the scope of the existing requirements covers the full spectrum of events that would be included in this rulemaking. The latter method of accomplishing this would have the potential advantage of leaving the requirements for command and control and organizational structures in a single regulation (i.e., 10 CFR part 50, appendix E, section IV.A). The NRC seeks stakeholder input on this subject.

VIII. Regulatory Flexibility Certification.

Under the Regulatory Flexibility Act (5 U.S.C. 605(b)), the NRC certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of “small entities” set forth in the Regulatory Flexibility Act or established in 10 CFR 2.810, “NRC size standards.”

IX. Availability of Regulatory Analysis.

The NRC has prepared a draft regulatory analysis and a draft supplemental regulatory analysis on this proposed regulation. The analyses examine the costs and benefits of the alternatives considered by the NRC. The NRC requests public comment on the draft regulatory analyses. The draft regulatory analyses are available as indicated in the “Availability of

Documents” section of this document. Comments on the draft analyses may be submitted to the NRC as indicated under the ADDRESSES section of this document.

X. Availability of Guidance.

The NRC is issuing for comment draft requlatory guidance to support the implementation of the proposed requirements in this rulemaking. You may access information and comment submissions related to the DGs by searching on http://www.regulations.gov under Docket ID NRC-2014-0240.

The DG-1301, “Flexible Mitigation Strategies for Beyond-Design-Basis Events,” provides licensees and applicants with an acceptable method of responding to an ELAP and demonstrating compliance with the proposed regulations requiring additional defense-in-depth measures for the mitigation of beyond-design-basis external events.

The DG-1317, “Wide-Range Spent Fuel Pool Level Instrumentation,” describes one method of addressing these challenges by providing safety enhancements in the form of reliable spent fuel pool instrumentation for beyond-design-basis external events.

The DG-1319, “Integrated Response Capabilities for Beyond-Design-Basis Events,” describes one method the NRC endorses to enhance a site’s ability to implement the on-site emergency preparedness programs and guidelines and better cope with conditions resulting from a beyond-design-basis external event.

You may submit comments on this draft regulatory guidance by the following methods:

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID NRC-2014-0240. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov.

- **Mail comments to:** Cindy Bladey, Office of Administration, Mail Stop: OWFN-12-H08, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

XI. Backfitting and Issue Finality.

Proposed Rule

As required by §§ 50.109, 52.63, 52.83, and 52.98, the Commission has completed a backfit and issue finality analysis for the proposed rule. The Commission finds that the backfits contained in the proposed rule, when considered in the aggregate, would constitute a substantial additional protection for defense-in-depth and are justified in view of this increased protection of the public health and safety. Availability of the backfit and issue finality analysis is indicated in the "Availability of Documents" section of this document.

Draft Regulatory Guidance

The NRC is issuing, for public comment, three DGs that would support implementation of this proposed rule: DG-1301, "Flexible Mitigation Strategies for Beyond-Design-Basis Events"; DG-1317, "Wide-Range Spent Fuel Pool Level Instrumentation"; and DG-1319, "Integrated Response Capabilities for Beyond-Design-Basis Events." These DGs would provide guidance on the methods acceptable to the NRC for complying with this proposed rule. The DGs would apply to all current holders of nuclear power plant construction permits and operating licensees (under 10 CFR part 50 and renewed licenses under 10 CFR part 54) and combined licenses under 10 CFR part 52, and applicants for construction permits; operating licensees; standard design certifications; standard design approvals; combined licenses that do not reference a standard design certification, standard design approval, or manufacturing license; and manufacturing licenses that do not reference a standard design certification or standard design approval.

Issuance of the DGs in final form would not constitute backfitting under § 50.109 and would not otherwise be inconsistent with the issue finality provisions in 10 CFR part 52. As discussed in the “Implementation” section of each DG, the NRC has no current intention to impose the DGs, if finalized, on current holders of a construction permit, operating license, or combined license.

Applying the DGs, if finalized, to applications for construction permits; operating licensees; standard design certifications; standard design approvals; combined licenses that do not reference a standard design certification, standard design approval, or manufacturing license; or manufacturing licenses that do not reference a standard design certification or standard design approval would not constitute backfitting as defined in § 50.109 or be otherwise inconsistent with the applicable issue finality provisions in 10 CFR part 52, because such applicants are not within the scope of entities protected by § 50.109 or the applicable issue finality provisions in 10 CFR part 52. Neither § 50.109 nor the issue finality provisions under 10 CFR part 52 – with certain exceptions – were intended to apply to every NRC action that substantially changes the expectations of current and future applicants. The exceptions to this principle for these DGs are: 1) a combined license applicant that references an already-issued standard design certification, standard design approval, or manufacturing license, which have their own specific issue finality provisions; and 2) a manufacturing license applicant that references an already-issued standard design certification or standard design approval. However, the proposed operational requirements addressed in the DGs are not within the scope of issues that may be resolved for design certification or design approval, and therefore are not afforded issue finality protection under 10 CFR part 52; whereas the design requirements addressed in the DGs are expected to be resolved during the design phase (e.g., design certification applications) and therefore would be afforded issue finality protection under 10 CFR part 52. Manufacturing licenses could contain operational matters that would be required by the proposed rule. If the NRC were to impose the DGs on such a licensee, the NRC would have to address the criteria for avoiding issue finality as described in the applicable issue finality provision.

XII. Cumulative Effects of Regulation.

The NRC engaged extensively with external stakeholders throughout this rulemaking and related regulatory activities. Public involvement has included: 1) issuance of two ANPRs and two draft regulatory basis documents that requested stakeholder feedback; 2) issuance of conceptual and preliminary proposed rule language in support of public meetings; 3) numerous public meetings with the ACRS; and 4) many more public meetings that supported both the development of the draft regulatory basis documents as well as development of the implementing guidance for the two orders that this rulemaking would make generically applicable (i.e., Orders EA-12-049 and EA-12-051). Section II.E of this notice provides a more detailed discussion of public involvement.

The NRC is following its CER process with regard to the issuance of draft guidance with the proposed rule to support more informed external stakeholder feedback. The "Availability of Guidance" section of this document describes how the public can access the draft guidance for which the NRC seeks external stakeholder feedback.

Finally, the NRC is requesting CER feedback on the following questions:

1. In light of the current or projected CER challenges, does the proposed rule's compliance dates provide sufficient time to implement the new proposed requirements, including changes to programs, procedures, and the facility? Specifically, the current proposed

rule would require each holder of an operating license or holder of a combined license for which the Commission made the finding specified in § 52.103(g) to comply with all provisions of the proposed rule no later than four years following the effective date of the rule, unless otherwise specified in proposed § 50.155 or proposed 10 CFR part 50, appendix E, section VII. The greatest implementation challenge may involve implementing plant-specific SAMGs that reflect the recent industry efforts to update the SAMGs, and reflect the new mitigation capabilities stemming from Order EA-12-049. This effort is not anticipated to require refueling outages to support implementation. The NRC requests feedback on what this time period should be, recognizing that licensee resources to implement SAMG requirements are likely also involved with implementation of the mitigation strategies for beyond-design-basis external events.

2. If current or projected CER challenges exist, what should be done to address this situation? For example if more time is required for implementation of the new requirements, what period of time would be sufficient?

3. Do other NRC regulatory actions, including the post-Fukushima actions and other any other actions (e.g., generic communications, license amendment requests, inspection findings of a generic nature), influence the implementation of the proposed rule's requirements?

4. Are there unintended consequences associated with implementation of these requirements, including implementing the requirements as a priority over other facility modifications that are currently being prioritized and scheduled?

5. Please provide feedback on the NRC's supporting regulatory analyses for this rulemaking. Of note, the regulatory analyses estimate the cost of implementing both

Order EA-12-049 and Order EA-12-051. The NRC would like to draw particular attention to the regulatory analysis estimates concerning SAMG requirements. The NRC would appreciate feedback regarding those estimates.

XIII. Plain Writing.

The Plain Writing Act of 2010 (Pub. L. 111-274) requires Federal agencies to write documents in a clear, concise, and well-organized manner. The NRC has written this document to be consistent with the Plain Writing Act as well as the Presidential Memorandum, "Plain Language in Government Writing," published June 10, 1998 (63 FR 31883). The NRC requests comment on this document with respect to the clarity and effectiveness of the language used.

XIV. Environmental Assessment and Proposed Finding of No Significant Environmental Impact.

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in subpart A of 10 CFR part 51, that this rule, if adopted, would not be a major Federal action significantly affecting the quality of the human environment, and an environmental impact statement is not required. The basis of this determination reads as follows: The proposed action will not result in any radiological effluent impact as it will not change any design basis structures, systems, or components that function to limit the release of radiological effluents during or after an accident. This proposed rule does not change the standards and requirements for radiological releases and effluents. None of the revisions or additions in the proposed rule would affect current occupational or public radiation exposure. The rule would not cause any significant non-radiological impacts, as it would not affect any historic sites or any non-radiological plant effluents. The NRC concludes that this proposed rule will not cause any significant radiological or non-radiological impacts on the human environment.

The determination of this environmental assessment is that there will be no significant effect on the quality of the human environment from this action. Public stakeholders should note, however, that comments on any aspect of this environmental assessment may be submitted to the NRC as indicated in the "Addresses" section of this document. The environmental assessment is available as indicated under the "Availability of Documents" section.

The NRC has sent a copy of the environmental assessment and this proposed rule to every State Liaison Officer and has requested comments.

XV. Paperwork Reduction Act.

This proposed rule contains new or amended information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq). This rule has been submitted to the OMB for approval of the information collection requirements.

Type of submission, new or revision: New

The title of the information collection: Mitigation of Beyond-Design-Basis Events Proposed Rule

The form number if applicable: Not applicable

How often the collection is required: Once and annually.

Who will be required or asked to report: Operating nuclear power reactor sites (comprised of 64 operating sites), as well as the Pressurized Water Reactor Owners Group and the Boiling Water Reactor Owners Group.

An estimate of the number of annual responses: 91

The estimated number of annual respondents: 66 (64 recordkeepers + 2 owners groups)

An estimate of the total number of hours needed annually to complete the requirement or request: 121,909

Abstract: In response to the Great East Japan Earthquake of March 11, 2011, the NRC is seeking to: 1) permanently write into agency rules the requirements already imposed by the Order EA-12-049 and Order EA-12-051; 2) create regulatory requirements for Severe Accident Management Guidelines (SAMGs) to promote consistency across industry; 3) incorporate emergency preparedness-related industry initiatives into the regulations; 4) provide requirements for mitigating strategies for new reactor designs; and 5) address a number of petitions for rulemaking (PRMs) submitted in the aftermath of the March 2011 Fukushima Dai-ichi event.

The NRC is seeking public comment on the potential impact of the information collections contained in this proposed rule and on the following issues:

1. Is the proposed information collection necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?

2. Is the estimate of burden accurate?

3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?

4. How can the burden of the information collection be minimized, including the use of automated collection techniques?

A copy of the OMB clearance package may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O-1 F21, Rockville, MD

20852. The OMB clearance package and rule are available at the NRC worldwide Web site: *http://www.nrc.gov/public-involve/doc-comment/omb/index.html* for 60 days after the signature date of this notice.

Send comments on any aspect of these proposed regulations related to information collections, including suggestions for reducing the burden and and on the previously stated issues, by **[INSERT DATE 30 DAYS AFTER PUBLICATION IN THE *FEDERAL REGISTER*]** to the Records and FOIA/Privacy Services Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by electronic mail to Infocollects.Resource@NRC.gov and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503. Comments on the proposed information collections may also be submitted via the Federal eRulemaking Portal http://www.regulations.gov, Docket No. NRC-2014-0240. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after this date.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

XVI. Criminal Penalties.

For the purposes of Section 223 of the Atomic Energy Act of 1954, as amended (AEA), the NRC is issuing this proposed rule that would amend 10 CFR parts 50 and 52 under one or more of Sections 161b, 161i, or 161o of the AEA. Willful violations of the rule would be subject to criminal enforcement. Criminal penalties as they apply to regulations in 10 CFR parts 50 and 52 are discussed in §§ 50.111 and 52.303.

XVII. Coordination with NRC Agreement States.

The Agreement States are receiving notification of the publication of this proposed rule.

XVIII. Compatibility of Agreement State Regulations.

Under the "Policy Statement on Adequacy and Compatibility of Agreement State Programs," approved by the Commission on June 20, 1997, and published in the Federal register (62 FR 46517; September 3, 1997), this rule is classified as compatibility "NRC." Compatibility is not required for Category "NRC" regulations. The NRC program elements in this category are those that relate directly to areas of regulation reserved to the NRC by the AEA or the provisions of Title 10 of the Code of Federal Regulations, and although an Agreement State may not adopt program elements reserved to the NRC, it may wish to inform its licensees of certain requirements via a mechanism that is consistent with a particular State's administrative procedure laws, but does not confer regulatory authority on the State.

XIX. Voluntary Consensus Standards.

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this proposed rule, the NRC will add requirements for the mitigation of beyond-design-basis events. This action does not constitute the establishment of a standard that contains generally applicable requirements.

XX. Public Meeting.

The NRC will conduct a public meeting on this proposed rule for the purpose of describing the proposed rule to the public and answering questions from the public on the proposed rule.

The NRC will publish a notice of the location, time, and agenda of the meeting in the *Federal Register*, on Regulations.gov, and on the NRC's public meeting Web site within at least 10 calendar days before the meeting. Stakeholders should monitor the NRC's public meeting Web site for information about the public meeting at: http://www.nrc.gov/public-involve/public-meetings/index.cfm.

XXI. Availability of Documents.

The documents identified in the following table are available to interested persons through one or more of the following methods, as indicated.

Document	ADAMS ACCESSION NO. / WEB LINK / FEDERAL REGISTER CITATION
Primary Rulemaking Documents	
Draft regulatory analysis and backfit and issue finality analysis	ML15049A212
Draft supplemental regulatory analysis	ML15069A278
Environmental assessment	ML15049A215
Draft Regulatory Guides	
DG-1301, Flexible Mitigation Strategies for Beyond-Design-Basis Events	ML13168A031
DG-1317, Wide-Range Spent Fuel Pool Level Instrumentation	ML14245A454
DG-1319, Integrated Response Capabilities for Beyond-Design-Basis Events	ML14265A070
Other References	
ACRS Transcript—Fukushima Full Committee, Discuss preliminary Mitigation of Beyond-Design-Basis Events Rulemaking Language, December 4, 2014	ML14345A387
ACRS Transcript—Fukushima Subcommittee, Discuss Preliminary Mitigation of Beyond-Design-Basis Events Rulemaking Language, November 21, 2014	ML14337A671

ACRS Transcript—Full Committee, Discuss Consolidation of Station Blackout Mitigation Strategies and Onsite Emergency Response Capabilities Rulemakings, July 10, 2014	ML14223A631
ACRS Transcript—Full Committee, Discuss the Station Blackout Mitigation Strategies Regulatory Basis, June 5, 2013	ML13175A344
ACRS Transcript—Joint Fukushima and PRA Subcommittee, Discuss CPRR Technical Analysis, August 22, 2014	ML14265A059
ACRS Transcript—Plant Operations and Fire Protection Subcommittee, Discuss the Onsite Emergency Response Capabilities Regulatory Basis, February 6, 2013	ML13063A403
ACRS Transcript—Reactor Safeguards Reliability and PRA Subcommittee, Discuss CPRR Technical Analysis, November 19, 2014	ML14337A651
ACRS Transcript—Regulatory Policies and Practices Subcommittee, Discuss the Station Blackout Mitigation Strategies regulatory basis, December 5, 2013 and April 23, 2013	ML13148A404
American National Standards Institute/American Nuclear Society 3.2-2012, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"	http://www.ans.org/store/
CLI-12-09, South Carolina Electric & Gas Co. and South Carolina Public Service Authority (Also Referred to as Santee Cooper)	ML12090A531
COMGBJ-11-0002, "NRC Actions Following the Events in Japan," March, 21, 2011	ML110800456
COMSECY-13-0002, "Consolidation of Japan Lessons Learned Near-Term Task Force Recommendations 4 and 7 Regulatory Activities," January 25, 2013	ML13011A037
COMSECY-13-0010, "Schedule and Plans for Tier 2 Order on Emergency Preparedness for Japan Lessons Learned," dated March 27, 2013	ML12339A262
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Throughout the development of this rule, the NRC may post documents related to this rule, including public comments, on the Federal rulemaking Web site at http://www.regulations.gov under Docket ID NRC-2014-0240. The Federal rulemaking Web site allows you to receive alerts when changes or additions occur in a docket folder. To subscribe: 1) Navigate to the docket folder NRC-2014-0240); 2) click the “Sign up for E-mail Alerts” link; and 3) enter your e-mail address and select how frequently you would like to receive e-mails (daily, weekly, or monthly).

List of Subjects

10 CFR Part 50

Antitrust, Classified information, Criminal penalties, Fire protection, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

10 CFR Part 52

Administrative practice and procedure, Antitrust, Backfitting, Combined license, Early site permit, Emergency planning, Fees, Inspection, Limited work authorization, Nuclear power plants and reactors, Probabilistic risk assessment, Prototype, Reactor siting criteria, Redress of site, Reporting and recordkeeping requirements, Standard design, Standard design certification, Incorporation by reference.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 552 and 553, the NRC is proposing to adopt the following amendments to 10 CFR parts 50 and 52.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for 10 CFR part 50 continues to read as follows:

Authority: Atomic Energy Act secs. 102, 103, 104, 105, 147, 149, 161, 181, 182, 183, 186, 189, 223, 234 (42 U.S.C. 2132, 2133, 2134, 2135, 2167, 2169, 2201, 2231, 2232, 2233, 2236, 2239, 2273, 2282); Energy Reorganization Act secs. 201, 202, 206 (42 U.S.C. 5841, 5842, 5846); Nuclear Waste Policy Act sec. 306 (42 U.S.C. 10226); Government Paperwork Elimination Act sec. 1704 (44 U.S.C. 3504 note); Energy Policy Act of 2005, Pub. L. No. 109-58, 119 Stat. 194 (2005). Section 50.7 also issued under Pub. L. 95-601, sec. 10, as amended by Pub. L. 102-486, sec. 2902 (42 U.S.C. 5851). Section 50.10 also issued under Atomic Energy Act secs. 101, 185 (42 U.S.C. 2131, 2235); National Environmental Protection Act sec. 102 (42 U.S.C. 4332). Sections 50.13, 50.54(d), 50.54(dd), and 50.103 also issued under Atomic Energy Act sec. 108 (42 U.S.C. 2138).

Sections 50.23, 50.35, 50.55, and 50.56 also issued under Atomic Energy Act sec. 185 (42 U.S.C. 2235). Appendix Q also issued under National Environmental Protection Act sec. 102 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415 (42 U.S.C. 2239). Section 50.78 also issued under Atomic Energy Act sec. 122 (42 U.S.C. 2152). Sections 50.80 - 50.81 also issued under Atomic Energy Act sec. 184 (42 U.S.C. 2234).

2. In § 50.8, paragraph (b) is revised to read as follows:

§ 50.8 Information collection requirements: OMB approval.

* * * * *

(b) The approved information collection requirements contained in this part appear in §§50.30, 50.33, 50.34, 50.34a, 50.35, 50.36, 50.36a, 50.36b, 50.44, 50.46, 50.47, 50.48, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.60, 50.61, 50.61a, 50.62, 50.63, 50.64, 50.65, 50.66, 50.68, 50.69, 50.70, 50.71, 50.72, 50.74, 50.75, 50.80, 50.82, 50.90, 50.91, 50.120, 50.150, 50.155, and appendices A, B, E, G, H, I, J, K, M, N, O, Q, R, and S to this part.

* * * * *

3. In § 50.34:

a. revise paragraphs (a)(13) and add paragraph (a)(14);

b. revise paragraph (b)(12) and add paragraph (b)(13); and

c. revise paragraph (i).

The revisions and additions read as follows:

§ 50.34 Contents of applications; technical information.

(a) * * *

(13) On or after July 13, 2009, power reactor applicants who apply for a construction permit shall submit the information required by 10 CFR 50.150(b) as a part of their preliminary safety analysis report.

(14) On or after **[EFFECTIVE DATE]**, power reactor applicants applying for a construction permit must submit a preliminary description of the design features included in the plant design under § 50.155(d), and an explanation of how those design features comply with the requirements of § 50.155(d).

(b) * * *

(12) On or after July 13, 2009, power reactor applicants who apply for an operating license which is subject to 10 CFR 50.150(a) shall submit the information required by 10 CFR 50.150(b) as a part of their final safety analysis report.

(13) On or after **[EFFECTIVE DATE]**, power reactor applicants applying for an operating license must submit a final description of the design features included in the plant design under § 50.155(d), and an explanation of how those design features comply with the requirements of § 50.155(d).

* * * * *

(i) *Mitigation of beyond-design-basis events*.

Each application for a power reactor operating license under this part must include the applicant's plans for implementing the requirements of § 50.155 and 10 CFR part 50, Appendix E, Section VII, including a schedule for achieving full compliance with these requirements. The application must also include a description of:

(1) the integrated response capability required by § 50.155(b);

(2) the equipment upon which the strategies and guidelines required by § 50.155(b)(1) rely, including the planned locations of the equipment and how the equipment and SSCs meet the design requirements of § 50.155(c); and

(3) the strategies and guidelines required by § 50.155(b)(2).

* * * * *

4. In § 50.54:

a. remove paragraph (hh)(2);

b. redesignate paragraph (hh)(3) as (hh)(2); and

c. revise paragraph (hh)(2) to read as follows:

§ 50.54 Conditions of licenses.

* * * * *

(hh) * * *

(2) This section does not apply to a licensee that has submitted the certifications required under § 50.82(a)(1) or § 52.110(a) of this chapter once the NRC has docketed those certifications.

5. Add § 50.155 to read as follows:

§ 50.155 Mitigation of Beyond-Design-Basis Events

(a) *Applicability*.

(1) Each holder of an operating license for a nuclear power reactor under this part and each holder of a combined license under part 52 of this chapter after the Commission has made the finding under § 52.103(g), before the NRC's docketing of the license holder's certifications described in section § 50.82(a)(1) or § 52.110(a) of this chapter, shall comply with the requirements of this section and section VII of appendix E to 10 CFR part 50.

(2) Each applicant for an operating license for a nuclear power reactor under this part and each holder of a combined license under part 52 of this chapter before the Commission has made the finding under § 52.103(g) shall comply with the requirements of this section and section VII of appendix E to 10 CFR part 50 no later than the date on which the Commission issues the operating license under § 50.57 or makes the finding under § 52.103(g), respectively.

(3) When the NRC has docketed the certifications described in § 50.82(a)(1) or § 52.110(a) of this chapter, submitted by a licensee subject to the requirements of this section and section VII of appendix E to 10 CFR part 50, then that licensee shall comply with the requirements of §§ 50.155(b)-(f) associated with maintaining or restoring secondary containment capabilities, if applicable, and spent fuel pool cooling capabilities, but not with § 50.155(c)(4) and section VII of appendix E to 10 CFR part 50, for the unit described in the § 50.82(a)(1) or § 52.110(a) certifications until the spent fuel pool(s) is empty of all irradiated fuel.

(i) Holders of operating licenses or combined licenses for which the NRC has docketed the certifications described in § 50.82(a)(1) or § 52.110(a) of this chapter need not meet the requirements of this section except for paragraph (b)(2) if the licensee performs and retains an analysis demonstrating that the decay heat of the fuel in the spent fuel pool is removed solely by heating and boiling of water within the spent fuel pool and the boil-off period provides sufficient time for the licensee to obtain off-site resources to sustain the spent fuel pool cooling function indefinitely.

(ii) Dominion Nuclear Connecticut, Inc. (Millstone Power Station Unit 1) is not subject to the requirements of this section.

(4) The requirements in paragraph (d) of this section apply to applicants for:

(i) Construction permits for nuclear power reactors under this part;

(ii) Operating licenses for nuclear power reactors under this part for which a construction permit was issued after [EFFECTIVE DATE];

(iii) Standard design certifications under part 52 of this chapter;

(iv) Standard design approvals under part 52 of this chapter;

(v) Combined licenses under part 52 of this chapter that do not reference a standard design certification, standard design approval, or manufacturing license; and

(vi) Manufacturing licenses under part 52 of this chapter that do not reference a standard design certification or standard design approval.

(b) *Integrated response capability*. Each applicant or licensee shall develop, implement, and maintain an integrated response capability that includes:

(1) Mitigation Strategies for Beyond-Design-Basis External Events.

Strategies and guidelines to mitigate beyond-design-basis external events from natural phenomena that result in an extended loss of all ac power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. These strategies and guidelines must be capable of being implemented site-wide and must include:

(i) Maintaining or restoring core cooling, containment, and spent fuel pool cooling capabilities; and

(ii) The acquisition and use of offsite assistance and resources to support the functions required by paragraph (b)(1)(i) of this section indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies.

(2) Extensive Damage Mitigation Guidelines (EDMGs).

Strategies and guidelines to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire, to include strategies and guidelines in the following areas:

(i) Firefighting;

(ii) Operations to mitigate fuel damage; and

(iii) Actions to minimize radiological release.

(3) Severe Accident Management Guidelines.

Strategies and guidelines for mitigating the consequences of events that progress to imminent or actual significant damage to fuel in the reactor vessel or spent fuel pool to support actions intended to:

(i) Arrest the progression of fuel damage;

(ii) Maximize the duration for which containment capability is maintained; and,

(iii) Minimize radiological releases.

(4) Integration of strategies and guidelines in paragraphs (b)(1) - (b)(3) of this section with the Emergency Operating Procedures (EOPs).

(5) Sufficient staffing to support implementation of the strategies and guidelines in paragraphs (b)(1) - (b)(3) of this section in conjunction with the EOPs to respond to events.

(6) A supporting organizational structure with defined roles, responsibilities, and authorities for directing and performing the strategies and guidelines in paragraphs (b)(1) – (b)(3) of this section.

(c) *Equipment*.

(1) The capacity and capability of the equipment relied on for the mitigation strategies required by paragraph (b)(1) of this section must be sufficient to simultaneously maintain or restore core cooling, containment, and spent fuel pool cooling capabilities for all the power reactor units within the site boundary.

(2) The equipment relied on for the mitigation strategies required by paragraph (b)(1) of this section must be reasonably protected from the effects of natural phenomena that are equivalent to the design basis of the facility.

(i) By **[DATE 730 DAYS AFTER THE EFFECTIVE DATE OF THE RULE]**, each licensee that received the March 12, 2012, NRC letter issued under § 50.54(f) concerning reevaluations of seismic and flooding hazard levels, shall provide reasonable protection against that reevaluated seismic or flooding hazard(s) if it exceeds the design basis of its facility.

(3) The equipment relied on for the mitigation strategies in paragraph (b)(1) of this section must receive adequate maintenance such that the equipment is capable of fulfilling its intended function.

(4) The equipment relied on for the mitigation strategies in paragraph (b)(1) of this section must include reliable means to remotely monitor wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions.

(d) *Design features*.

Each applicant listed in paragraph (a)(4) shall include design features in the plant design sufficient to enhance coping durations and minimize reliance on human actions to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities during an extended loss of all ac power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink.

(e) *Training requirements*.

Each licensee shall provide for the training and qualification of personnel that perform activities in accordance with the strategies and guidelines identified in paragraphs (b)(1)-(b)(3) of this section. The training and qualification on these activities must be developed using the systems approach to training as defined in § 55.4 except for elements already covered under other NRC regulations.

(f) *Drills and Exercises*.

(1) An applicant for an operating license issued under this part shall conduct an initial drill or exercise that demonstrates the capability to transition to and use one or more of the strategies and guidelines in paragraphs (b)(1)-(b)(3) of this section and use the communications equipment required in 10 CFR part 50, appendix E, section VII no more than 12 months before issuance of an operating license for the unit described in the license application.

(2) A holder of a combined license issued under 10 CFR part 52 before the Commission has made the finding under § 52.103(g), shall conduct an initial drill or exercise that demonstrates the capability to transition to and use one or more of the strategies and guidelines in paragraphs (b)(1)-(b)(3) of this section and use the communications equipment required in 10 CFR part 50, appendix E, section VII no more than 12 months before the date specified for completion of the last inspections, tests, and analyses in the ITAAC completion schedule required by § 52.99(a) for the unit described in the combined license.

(3) Once the Commission issues an operating license to an entity described in paragraph (f)(1) of this section or makes the finding under § 52.103(g) for an entity described in paragraph (f)(2) of this section, the licensee shall conduct subsequent drills, exercises, or both that collectively demonstrate a capability to use at least one of the strategies and guidelines in each of paragraphs (b)(1)-(b)(3) of this section in succeeding 8-year intervals. The drills and exercises performed to demonstrate this capability must include transitions from other procedures and guidelines as applicable, and the use of communications equipment required in 10 CFR part 50, appendix E, section VII. Each licensee shall not exceed 8 years between any consecutive drills or exercises.

(4) A holder of an operating license issued under this part or a combined license under 10 CFR part 52 for which the Commission has made the finding specified in § 52.103(g) as of **[EFFECTIVE DATE]**, shall conduct an initial drill or exercise that demonstrates the capability to transition to and use one or more of the strategies and guidelines in paragraphs (b)(1)-(b)(3) of this section and use communications equipment required in 10 CFR part 50, appendix E, section VII by **[DATE 4 YEARS AFTER EFFECTIVE DATE]**. Following this initial drill or exercise, the licensee shall conduct subsequent drills, exercises, or both that collectively demonstrate a capability to use at least one of the strategies and guidelines in each of paragraphs (b)(1)-(b)(3) of this section in succeeding 8-year intervals. The drills and exercises

performed to demonstrate this capability must include transitions from other procedures and guidelines as applicable, and the use of communications equipment required in 10 CFR part 50, appendix E, section VII. Each licensee shall not exceed 8 years between any consecutive drills or exercises.

(g) *Change Control*.

(1) A licensee may make changes in the implementation of the requirements in this section and 10 CFR part 50, appendix E, section VII without NRC approval, *provided* that before implementing each such change, the licensee performs an evaluation demonstrating that the provisions of this section and 10 CFR part 50, appendix E, section VII continue to be met.

(2) Documentation of all changes, including the evaluation required by paragraph (g)(1) of this section, shall be maintained until the requirements of this section and section VII of appendix E to 10 CFR part 50 no longer apply.

(3) Changes in the implementation of requirements in this chapter subject to change control processes other than paragraph (g) of this section and resulting from changes in the implementation of the requirements in this section and 10 CFR part 50, appendix E, section VII must be processed via their respective change control processes.

(4) For a licensee of a nuclear power reactor who was subject to paragraph (d) of this section as an applicant, if the licensee makes changes to the design features included in the plant design under paragraph (d) of this section, then the licensee shall ensure that the changed design feature continues to satisfy the design objectives described in paragraph (d) of this section.

(h) *Implementation.*

Unless otherwise specified in this section or 10 CFR part 50, appendix E, section VII:

(1) Each holder of an operating license under this part on **[EFFECTIVE DATE OF THE FINAL RULE]** shall comply with all the provisions of this section no later than four years following **[EFFECTIVE DATE OF THE FINAL RULE]**.

(2) Each holder of a combined license under 10 CFR part 52 for which the Commission made the finding specified in § 52.103(g) as of **[EFFECTIVE DATE OF THE FINAL RULE]** shall comply with all the provisions of this section no later than four years following **[EFFECTIVE DATE OF THE FINAL RULE]**.

6. In Appendix E to part 50:

a. revise paragraph I.2;

b. revise paragraphs IV.B.1, IV.E.2, and IV.F.2.j;

c. revise paragraph VI.3.c; and

d. add Section VII.

The revisions and additions read as follows:

Appendix E to Part 50—Emergency Planning and Preparedness for Production and Utilization Facilities

* * * * *

I. * * *

2. This appendix establishes minimum requirements for emergency plans for use in attaining an acceptable state of emergency preparedness. These plans shall be described generally in the preliminary safety analysis report for a construction permit and submitted as part of the final safety analysis report for an operating license. These plans, or major features thereof, may be submitted as part of the site safety analysis report for an early site permit.

Section VII of this appendix also provides for “Communications and Staffing Requirements for the Mitigation of Beyond-Design-Basis Events” that do not need to be contained within a licensee’s emergency plan.

* * * * *

IV. * * *

B. * * *

1. The means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials, including from all reactor core and spent fuel pool sources, shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. By June 20, 2012, for nuclear power reactor licensees, these action levels must include hostile action that may adversely affect the nuclear power plant. The initial emergency action levels shall be discussed and agreed on by the applicant or licensee and state and local governmental authorities, and approved by the NRC. Thereafter, emergency action levels shall be reviewed with the State and local governmental authorities on an annual basis.

* * * * *

E. * * *

2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials, including from all reactor core and spent fuel pool sources, to the environment;

*   *   *   *   *

F.   *   *   *

2.   *   *   *

j. The exercises conducted under paragraph 2 of this section by nuclear power reactor licensees must provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to implement the principal functional areas of emergency response identified in paragraph 2.b of this section. Each exercise must provide the opportunity for the ERO to demonstrate key skills specific to emergency response duties in the control room, TSC, OSC, EOF, and joint information center. Additionally, in each eight calendar year exercise cycle, nuclear power reactor licensees shall vary the content of scenarios during exercises conducted under paragraph 2 of this section to provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to respond to the following scenario elements: hostile action directed at the plant site, no radiological release or an unplanned minimal radiological release that does not require public protective actions, an initial classification of or rapid escalation to a Site Area Emergency or General Emergency, and integration of offsite resources with onsite response. The licensee shall maintain a record of exercises conducted during each eight year exercise cycle that documents the content of scenarios used to comply with the requirements of this paragraph. Each licensee shall conduct a hostile action exercise for each of its sites no later than December 31, 2015. The first eight-year exercise cycle for a site will begin in the calendar year in which the first hostile action exercise is conducted. For a site licensed under Part 52, the first eight-year exercise cycle begins in the calendar year of the initial exercise required by Section IV.F.2.a.

*   *   *   *   *

VI.   *   *   *

3.   *   *   *

c. In the event of a failure of NRC-supplied equipment, a replacement will be furnished by the NRC for licensee installation.

*   *   *   *   *

Section VII. Communications and Staffing Requirements for the Mitigation of Beyond Design Basis Events

All changes associated with implementation of the requirements in this section are subject to § 50.155(g). The change control provisions of § 50.54(q) do not apply to proposed changes associated with implementation of the requirements in this section, unless the requirements in this section are implemented within the licensee's emergency plan.

1. Each nuclear power reactor applicant or licensee shall perform a detailed analysis demonstrating that sufficient staff is available to implement the guidelines and strategies to respond to a beyond design basis external event resulting in impeded access to the nuclear power plant, an extended loss of ac power sources concurrent with either a loss of normal access to ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink, and affecting all units on-site.

a. An applicant for a power reactor operating license under this part shall perform this analysis at least 2 years before the issuance of the first operating license for full power (one authorizing operation above 5 percent of rated thermal power).

b. A holder of a combined license issued under 10 CFR part 52 before the Commission has made the finding under § 52.103(g) shall perform this analysis and submit it to the NRC under § 52.3 at least two years before the date specified for completion of the last inspections, tests, and analyses in the ITAAC completion schedule required by § 52.99(a) for the plant.

c. Each holder of a power reactor operating license or combined license for which the Commission has made the finding specified in § 52.103(g) as of **[EFFECTIVE DATE]** shall, before the NRC's docketing of the license holder's certifications described in section § 50.82(a)(1) or § 52.110(a) of this chapter, perform this analysis and submit it to the NRC under § 50.4 no later than **[DATE 365 DAYS AFTER EFFECTIVE DATE]**.

2. Each nuclear power reactor applicant or licensee shall make and describe adequate provisions for at least one onsite and one offsite communications system capable of remaining functional during an extended loss of alternating current power including the effects of the loss of the local communications infrastructure.

a. An applicant for a power reactor operating license under this part shall make these provisions no later than the issuance of the first operating license for full power (one authorizing operation above 5 percent of rated thermal power).

b. A holder of a combined license issued under 10 CFR part 52 before the Commission has made the finding under § 52.103(g) shall make these provisions no later than the date specified for completion of the last inspections, tests, and analyses in the ITAAC completion schedule required by § 52.99(a) for the plant.

c. Each holder of a power reactor operating license under this part or a combined license issued under 10 CFR part 52 for which the Commission has made the finding specified in § 52.103(g) as of **[EFFECTIVE DATE]** shall, before the NRC's docketing of the license holder's certifications described in section § 50.82(a)(1) or § 52.110(a) of this chapter, make these provisions no later than **[DATE 365 DAYS AFTER EFFECTIVE DATE]**.

Part 52 – LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

7. The authority citation for part 52 continues to read as follows:

Authority: Atomic Energy Act secs. 103, 104, 147, 149, 161, 181, 182, 183, 185, 186, 189, 223, 234 (42 U.S.C. 2133, 2201, 2167, 2169, 2232, 2233, 2235, 2236, 2239, 2282); Energy Reorganization Act secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); Government Paperwork Elimination Act sec. 1704 (44 U.S.C. 3504 note); Energy Policy Act of 2005, Pub. L. 109-58, 119 Stat. 594 (2005).

8. In § 52.47, add new paragraph (a)(29) to read as follows:

§ 52.47 Contents of applications; technical information.

* * * * *

(a) * * *

(29) For an application for a standard design certification that is subject to 10 CFR 50.155(d), a description of the design features included in the plant design under § 50.155(d) and an explanation of how those design features comply with the requirements of § 50.155(d).

* * * * *

9. In § 52.79, add new paragraph (a)(48) to read as follows:

§ 52.79 Contents of applications; technical information in final safety analysis report.

* * * * *

(a) * * *

(48) For an application for a combined license that is subject to § 50.155(d), a description of the design features included in the plant design under § 50.155(d) and an explanation of how those design features comply with the requirements of § 50.155(d).

* * * * *

10. In § 52.80, revise paragraph (d) to read as follows:

§ 52.80 Contents of applications; additional technical information.

* * * * *

(d)The applicant's plans for implementing the requirements of § 50.155 and 10 CFR part 50, appendix E, section VII, including a schedule for achieving full compliance with these requirements, and a description of:

(1) the integrated response capability required by § 50.155(b);

(2) the equipment upon which the strategies and guidelines required by § 50.155(b)(1) rely, including the planned locations of the equipment and how the equipment and SSCs meet the design requirements of § 50.155(c); and

(3) the strategies and guidelines required by § 50.155(b)(2).

11. In § 52.137, add new paragraph (a)(27) to read as follows:

§ 52.137 Contents of applications; technical information.

*	*	*	*	*

(a)	*	*	*

(27) For an application for a standard design approval that is subject to § 50.155(d), a description of the design features included in the plant design under § 50.155(d) and an explanation of how those design features comply with the requirements of § 50.155(d).

12. In § 52.157, add new paragraph (f)(33) to read as follows:

§ 52.157 Contents of applications; technical information in final safety analysis report.

*	*	*	*	*

(f)	*	*	*

(33) For an application for a manufacturing license that is subject to § 50.155(d), a description of the design features included in the plant design under § 50.155(d) and an explanation of how those design features comply with the requirements of § 50.155(d).

Dated at Rockville, Maryland, this xxth day of Xxxxx, 2015.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,
Secretary of the Commission.

§ 52.157 Contents of applications; technical information in final safety analysis report.

* * * * *

(f) * * *

(33) For an application for a manufacturing license that is subject to § 50.155(d), a description of the design features included in the plant design under § 50.155(d) and an explanation of how those design features comply with the requirements of § 50.155(d).

Dated at Rockville, Maryland, this xxth day of Xxxxx, 2015.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,
Secretary of the Commission.

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OFFICE	NRR/DPR/PRMB/PM	NRR/DPR/PRMB/RS	NRR/JLD/PM	NRR/DPR/PRMB/BC	NRR/DPR/DD	NRR/DPR/D
NAME	TReed	GLappert	EBowman	TInverso	AMohseni	LKokajko
DATE	2/26/2015	2/26/2015	2/26/2015	02/20/2015	2/26/2015	3/5/2015
OFFICE	NRO	NRR/JLD/D*	OIS/IRSD/TL*	OE/D*	NRO/D*	NSIR/D*
NAME	JShea	JDavis	TDonnell	PHolahan (KHanley for)	GTracy	JWiggins
DATE	Non-concurred	3/10/2015	3/31/2015	3/30/2015	3/31/2015	03/27/2015
OFFICE	ADM/TE*	NMSS/D*	RES/D*	CFO*	OGC*	NRR/D
NAME	CBladey	CHaney	BSheron (SCoffin for)	MWylie (TChampion for)	MSpencer	WDean
DATE	04/13/2015	3/27/2015	4/01/2015	3/30/2015	04/10/2015	04/'16/2015
OFFICE	EDO					
NAME	MSatorius					
DATE						

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