NRC Staff Review of National Academy of Sciences Report, "Lessons Learned from the Fukushima Dai-ichi Nuclear Accident for Improving Safety of U.S. Nuclear Plants"

Overview

In the Energy and Water Development Appropriations Act of 2012, Congress mandated that the U.S. Nuclear Regulatory Commission (NRC) fund a National Academy of Sciences (NAS) study of the lessons learned from the accident at the Fukushima Dai-ichi nuclear plant.

Immediately after the accident, nuclear plant operators and regulators around the world promptly began initiatives to determine if changes were needed to cope with extreme natural events.

In the U.S., the nuclear industry began an effort to understand the accident and identify appropriate safety enhancements. The NRC initiated a parallel effort to reevaluate nuclear plant safety regulations in light of the accident. These efforts were leveraged by insights from similar ongoing industry, government, and international activities. The NRC established the Near-Term Task Force (NTTF) to review NRC processes and regulations and make recommendations to improve them. The NRC subsequently created the Japan Lessons-Learned Project Directorate, which later consolidated with a newly established Mitigating Strategies Directorate to form a division within the Office of Nuclear Reactor Regulation. Established in June 2014, the Japan Lessons-Learned Division manages implementation and assessment of NTTF recommendations and other tiered actions. Several of the task force's recommendations were completed while the NAS was performing its study and work was ongoing on others.

The NAS study is one of many performed in the four years since the accident. The committee did not perform an evaluation of the industry, government, and international efforts, but used their written products to inform its work. The study was intended to be a broad-scope and high-level review to identify possible improvements to the safety and security of U.S. nuclear plants, taking into account, where possible, the results of other investigations and assessments.

NAS released their report, titled "Lessons Learned from the Fukushima Dai-ichi Nuclear Accident for Improving Safety of U.S. Nuclear Plants," in July 2014, documenting nine findings and 10 recommendations, which are discussed below. Findings are statements of NAS observations that form the basis of NAS recommendations. For each finding that has one or more corresponding recommendations, the recommendations were compared to ongoing and planned NRC and industry activities.

The focus of the staff's assessment was to determine if these ongoing or planned NRC and industry actions adequately address NAS's recommendations. The staff found that all NAS's recommendations are being adequately addressed. This enclosure documents NRC and industry activities that address the NAS recommendations.

As a matter of clarification, in the Statement of Task for the study, NAS stated that the National Research Council will provide an assessment of lessons learned from the Fukushima Dai-ichi nuclear accident for improving the safety and security of nuclear plants in the U.S. NAS notes in Section 1.3 of their study, "Strategy to Address the Study Charge," that with one exception, which deals with the loss of power to the security systems at Fukushima Dai-ichi, the security portion of the study task is not addressed in this report.

NAS Findings, Recommendations, and Staff Assessment

Finding 3.1:

The overarching lesson learned from the Fukushima Dai-ichi accident is that nuclear plant licensees and their regulators must actively seek out and act on new information about hazards that have the potential to affect the safety of nuclear plants.

Specifically:

1. Licensees and their regulators must continually seek out new scientific information about nuclear plant hazards and methodologies for estimating their magnitudes, frequencies, and potential impacts;

2. Nuclear plant risk assessments must incorporate these new information and methodologies as they become available; and

3. Plant operators and regulators must take timely actions to implement countermeasures when such new information results in substantial changes to risk profiles at nuclear plants.

Finding 4.1:

The accident at the Fukushima Dai-ichi nuclear plant was initiated by the March 11, 2011, Great East Japan Earthquake and the resulting tsunami. The earthquake knocked out offsite alternating current (AC) power to the plant and the tsunami inundated portions of the plant site. Flooding of critical plant equipment resulted in the extended loss of onsite AC and direct current (DC) power with the consequent loss of reactor monitoring, control, and cooling functions in multiple units. Three reactors sustained severe core damage (Units 1, 2, and 3); three reactor buildings were damaged by hydrogen explosions (Units 1, 3, and 4); and offsite releases of radioactive materials contaminated land in Fukushima Dai-ichi and several neighboring prefectures. The accident prompted widespread evacuations of local populations and distress of the Japanese citizenry, large economic losses, and the eventual shutdown of all nuclear power plants in Japan.

Personnel at the Fukushima Dai-ichi plant responded with courage and resilience during the accident in the face of harsh circumstances; their actions likely reduced the severity of the accident and the magnitude of offsite radioactive material releases. Several factors prevented plant personnel from achieving greater success—in particular averting reactor core damage—and contributed to the overall severity of the accident:

1. Failure of the plant owner (Tokyo Electric Power Company) and the principal regulator (Nuclear and Industrial Safety Agency) to protect critical safety equipment at the plant from flooding, in spite of mounting evidence that the plant's current design basis for tsunamis was inadequate.

2. The loss of nearly all onsite AC and DC power at the plant—with the consequent loss of real-time information for monitoring critical thermodynamic parameters in reactors, containments, and spent fuel pools and for sensing and actuating critical valves and equipment—greatly narrowed options for responding to the accident.

3. As a result of (1) and (2), the Unit 1, 2, and 3 reactors were effectively isolated from their ultimate heat sink (the Pacific Ocean) for a period of time far in excess of the heat capacity of the suppression pools or the coping time of the plant to station blackout.

4. Multi-unit interactions complicated the accident response. Unit operators competed for physical resources and the attention and services of staff in the onsite emergency response center.

5. Operators and onsite emergency response center staff lacked adequate procedures and training for accidents involving extended loss of all onsite AC and DC power, particularly procedures and training for managing water levels and pressures in reactors and their containments and hydrogen generated during reactor core degradation.

6. Failures to transmit information and instructions in an accurate and timely manner hindered responses to the accident. These failures resulted partly from the loss of communications systems and the challenging operating environments throughout the plant.

7. The lack of clarity of roles and responsibilities within the onsite emergency response center and between the onsite and headquarters emergency response centers may have contributed to response delays.

8. Staffing levels at the plant were inadequate for managing the accident because of its scope (affecting several reactor units) and long duration.

Finding 5.1:

Nuclear plant operators and regulators in the U.S. and other countries have identified and are taking useful actions to upgrade nuclear plant systems, operating procedures, and operator training in response to the Fukushima Dai-ichi accident. In the U.S., these actions include the nuclear industry's diverse and flexible coping strategies (FLEX) initiative as well as regulatory changes proposed by the NTTF. Implementation of these actions is still underway; consequently, it is too soon to evaluate their comprehensiveness, effectiveness, or status in the regulatory framework.

Based on the above findings, the committee made the following recommendations:

Recommendation 5.1A:

As the nuclear industry and its regulator implement the actions referenced in Finding 5.1, they should give specific attention to improving plant systems in order to enable effective responses to beyond-design-basis events, including, when necessary, developing and implementing ad hoc responses to deal with unanticipated complexities.

Attention to availability, reliability, redundancy, and diversity of plant systems and equipment is specifically needed for:

1. DC power for instrumentation and safety system control.

2. Tools for estimating real-time plant status during loss of power.

3. Decay-heat removal and reactor depressurization and containment venting systems and protocols.

4. Instrumentation for monitoring critical thermodynamic parameters in reactors, containments, and spent fuel pools.

5. Hydrogen monitoring (including monitoring in reactor buildings) and mitigation.

6. Instrumentation for both onsite and offsite radiation and security monitoring.

7. Communications and real-time information systems to support communication and coordination between control rooms and technical support centers, control rooms and the field, and between onsite and offsite support facilities.

Recommendation addressed by:

The Fukushima Dai-ichi accident prompted the NRC and the U.S. nuclear industry to take several actions to better understand and develop strategies to mitigate the risks from beyond-design-basis events. These actions include upgrading and augmenting plant systems, enhancing operating procedures, and improving operator training to enable more effective responses. A discussion of each of those actions follows.

<u>Mitigation Strategies Order EA-12-049</u>

On March 12, 2012, the NRC issued Order EA-12-049, “Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events.” The order requires a three-phase approach to mitigate beyond-design-basis external events. The initial phase requires using installed equipment and resources to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling capabilities. The transition phase requires providing sufficient portable onsite equipment and consumables to maintain or restore these functions until they can be performed with resources external to the site (i.e., offsite). The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely.

Nuclear Energy Institute (NEI) 12-06, “Diverse and Flexible Coping Strategies Implementation Guide,” Revision 0, outlines an approach for adding diverse and flexible mitigation strategies—or FLEX—that will increase defense in depth for beyond-design-basis scenarios to handle an extended loss of AC power (ELAP) and loss of normal access to the ultimate heat sink occurring simultaneously at all units on a site. NRC staff endorsed this guidance as an acceptable approach to meeting the requirements of the Mitigation Strategies Order.

FLEX consists, in part, of the following elements:

- Portable equipment that provides means of obtaining power and water to maintain or restore key safety functions for all reactors at a site. This could include equipment such as portable pumps, generators, batteries and battery chargers, compressors, hoses, couplings, tools, debris clearing equipment, temporary flood protection equipment and other supporting equipment or tools.

- Reasonable staging and protection of portable equipment from beyond-design-basis external events, such as earthquakes and floods, applicable to a site. The equipment used for FLEX would be staged and reasonably protected from applicable site-specific severe external events to provide reasonable assurance that N sets of FLEX equipment will remain deployable following such an event, where N is the number of units on site.

- Procedures and guidance to put FLEX strategies into place. FLEX Support Guidelines (FSGs), to the extent possible, will provide pre-planned FLEX strategies for accomplishing specific tasks to support emergency operating procedure and abnormal operating procedure functions to improve the capability to cope with beyond-design-basis external events.

- Programmatic controls that assure the continued viability and reliability of the FLEX strategies. These controls would establish standards for quality, maintenance, testing of FLEX equipment, configuration management, and periodic training of personnel.

NEI 12-06 addresses the first element of Recommendation 5.1A, DC power, in Section 3.2.2, Guideline (6), which specifies: "Plant procedures/guidance should identify loads that need to be stripped from the plant dc buses (both Class 1E and non-Class 1E) for the purpose of conserving DC power."

NEI 12-06 addresses the second element of Recommendation 5.1A, tools for estimating real-time plant status during a loss of power, in Section 5.3.3, Paragraph 1, which specifies that:

> [E]ach plant should compile a reference source for the plant operators that provides approaches to obtaining necessary instrument readings to support the implementation of the coping strategy (see Section 3.2.1.10). This reference source should include control room and non-control room readouts and should also provide guidance on how and where to measure key instrument readings at containment penetrations, where applicable, using a portable instrument (e.g., a Fluke meter). Such a resource could be provided as an attachment to the plant procedures/guidance. Guidance should include critical actions to perform until alternate indications can be connected and on how to control critical equipment without associated control power.

NEI 12-06 addresses the third element of Recommendation 5.1A, decay-heat removal and reactor depressurization and containment venting systems and protocols, in several sections. Section 3.2.2, Guideline (2), specifies that plant procedures/guidance should recognize the importance of steam-driven systems, such as auxiliary feed water, high-pressure coolant injection, reactor core isolation cooling, and passive systems, such as the isolation condenser, during the early stages of the event and direct the operators to invest appropriate attention to assuring its initiation and continued, reliable operation throughout the transient since this ensures decay heat removal. Reactor depressurization and containment venting systems and protocols are specific strategies listed in NEI 12-06 for the boiling-water (BWR) and pressurized-water reactors (PWR) and are elements of the strategies put in place by U.S. licensees in their plans.

NEI 12-06 addresses the fourth element of Recommendation 5.1A, instrumentation, in Section 3.2.1.10, which specifies that:

> Actions specified in plant procedures/guidance for loss of ac power are predicated on use of instrumentation and controls powered by station batteries. In order to extend battery life, a minimum set of parameters necessary to support strategy implementation should be defined. The parameters selected must be able to demonstrate the success of the strategies at maintaining the key safety functions as well as indicate imminent or actual core damage to facilitate a decision to manage the response to the event within the Emergency Operating Procedures and FLEX Support Guidelines or within the SAMGs [Severe Accident Management Guidelines].

NEI 12-06 addresses the fifth element of Recommendation 5.1A, hydrogen mitigation, in Section 3.2.2, Guideline (15), which specifies that "Procedures/guidance for units with BWR Mark III and PWR Ice Condenser containments should address the deployment of portable power supplies for providing backup power to the containment hydrogen igniters, including a prioritization approach for deployment." The NRC is evaluating the need for additional hydrogen mitigation measures as part of ongoing activities related to the lessons learned from the Fukushima Dai-ichi accident under NTTF Recommendation 6.

The sixth element of Recommendation 5.1A, instrumentation for both onsite and offsite radiation and security monitoring, is addressed in a number of documents. The fundamental requirements for onsite and offsite radiation monitoring equipment are addressed in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.47, "Emergency Plans," and Appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities." NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communication Capabilities," Section 3.5.1, provides guidance for licensees to assess the number of radiation protection technicians that would be required following a beyond design basis external event. NEI 12-06, Section 12.2 provides guidance for licensees to assess the need for additional radiation protection equipment, such as survey instruments, dosimetry, and equipment for off-site monitoring/sampling for the purpose of off-site and on-site radiological monitoring for an event lasting more than 24 hours.

It should also be noted that during an event, additional monitoring would be provided through the Nuclear/Radiological Incident Annex (NRIA) to the National Response Framework (NRF), which describes the policies, situations, concepts of operations, and responsibilities of the Federal departments and agencies governing the immediate response and short-term recovery activities for incidents involving release of radioactive materials to address the consequences of the event. There is a Tier 3 action for NRC to work with FEMA, States, and other external stakeholders to evaluate insights from the Fukushima Dai-ichi accident to identify potential enhancements to the U.S. decision-making framework.

In terms of instrumentation for monitoring security, 10 CFR 73.55(i) requires that licensee's establish and maintain intrusion detection and assessment systems that provide, at all times, the capability to detect and assess unauthorized persons and facilitate the effective implementation of the licensee's protective strategy. Included in that requirement is the requisite to ensure that the systems remain operable from an uninterruptible power supply in the event of the loss of normal power. Additionally, as it pertains to implementation of FLEX strategies, NEI 12-06, Section 3.2.2, Guideline (9), provides guidance concerning the possible adverse effects the loss of the preferred or Class 1E power supplies may have on security systems, possibly preventing area to access into the protected area and internal locked areas where remote equipment operation may be necessary.

NEI 12-01 addresses the seventh element of Recommendation 5.1A, communications and real-time information systems to support communication and coordination between control rooms and technical support centers, control rooms and the field, and between onsite and offsite support facilities. NEI 12-01 provides NRC accepted guidance to licensees for the assessment of current communications systems and equipment used during an emergency; consideration of enhancements that may be appropriate for the emergency plan with respect to communications requirements of 10 CFR 50.47 and Appendix E to 10 CFR 50; and consideration of the means necessary to power the new and existing communications equipment during a multi-unit event, with a loss of all AC power.

Also, the FLEX strategies were developed with ad-hoc responses to unanticipated complexities in mind. Specifically, operators are provided with the equipment, training, and procedures to allow them to efficiently respond to any scenario, including those that aren't anticipated. Additional equipment, if needed, is available from other sites and the National SAFER Response Centers; more personnel, if needed, are available through the Institute of Nuclear Power Operations (INPO) through pre-established mechanisms.

NRC staff reviews the licensees' overall integrated plans (OIPs) for compliance with the Mitigation Strategies Order, to verify that the plans, if carried out as described, would offer a reasonable path to comply with the order. The NRC staff conducts post-compliance inspections after all units at a site indicate compliance with the order.

Spent Fuel Pool Instrumentation Order EA-12-051

On March 12, 2012, the NRC issued Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," requiring all U.S. nuclear power plants to install reliable water level measurement instrumentation in their SFPs. The instrumentation must remotely report at least three distinct water levels: (1) normal level, (2) low level but still high enough to shield workers above the pools from radiation, and (3) a very low level near the top of the spent fuel rods (indicating that more water should be added without delay).

In the course of developing Order EA-12-051, the NRC considered requiring instruments for SFP water temperature and area radiation levels, as the NTTF had recommended. In light of the underlying technical justification for requiring the reliable level instrumentation, avoiding distracting resources from addressing problems with the reactor plant itself, which can progress more rapidly, the NRC concluded that these extra sensors would not be necessary. SFPs at reactor facilities in the U.S. are licensed to rely on heat removal through boiling water within the SFP. The loss of water within the SFP occurs through boil-off at a slow rate such that the level instrumentation alone can supply sufficient information for the operators to prioritize resources to mitigate events.

The NRC staff reviews the licensees' OIPs for installing reliable water-level measurement instrumentation to verify that the plans, if carried out as described, would offer a reasonable path to compliance with the order. The NRC staff will conduct post-compliance inspections after all units at a site indicate compliance with the order.

The NRC also examined the risks and consequences of postulated SFP accidents in "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor." The study offers publicly available consequence estimates of a low-likelihood seismic event at a specific reference plant causing a hypothetical SFP accident. This study's results are consistent with earlier research conclusions that SFPs are robust structures that are likely to withstand severe earthquakes without leaking. Based on this study and previous studies, the NRC continues to believe that SFPs provide adequate protection of public health and safety.

Reliable Hardened Containment Vents for Boiling-Water Reactors with Mark I and II Designs (Order EA-12-050 and Order EA-13-109)

The third element of Recommendation 5.1A, decay-heat removal and containment venting system, is also addressed by Order EA-12-050, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents," that the NRC issued on March 12, 2012, requiring all operating BWRs in the U.S. with Mark I and Mark II containments to install a reliable hardened vent. After issuing the order, other NRC evaluations examined the benefits of venting after reactor core damage occurs. SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems," was submitted to the Commission on November 26, 2012. In the staff requirements memorandum (SRM) for SECY-12-0157, the staff was directed to require BWR licensees with Mark I and Mark II containments to "upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions." On June 6, 2013, the staff issued Order EA-13-109, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions." The order includes additional requirements to ensure that the vents will remain functional in the conditions that would exist following reactor core damage.

The requirements of Order EA-13-109 are intended to increase the reliability of BWR Mark I and II containment venting to support decay heat removal from the reactor core and to provide protection against over-pressurization of the primary containments before core damage.

The revised order contains two distinct phases of implementation. Phase 1, which must be put in place by June 2018, requires licensees to upgrade the venting capabilities from the containment wetwell to provide reliable hardened vents that would remain functional during severe accident conditions. Phase 2, which all licensees are required to put in place by June 2019, requires licensees to: (a) install a reliable severe-accident-capable drywell vent system, or (b) develop a reliable containment venting strategy that makes it unlikely to need to vent from the containment drywell during severe accident conditions.

Containment Protection and Release Reduction Rulemaking

The third element of Recommendation 5.1A, decay-heat removal, reactor depressurization, and containment venting systems, is also addressed by the ongoing Containment Protection and Release Reduction (CPRR) rulemaking. This rulemaking addresses BWR Mark I and Mark II containment failure prevention measures, measures for controlling releases through the containment venting systems, and external water addition into the reactor pressure vessel or the drywell.

In the SRM for SECY-12-0157, the Commission directed the NRC staff to develop the regulatory basis in support of a rulemaking for filtering strategies with drywell filtration and severe accident management of BWRs with Mark I and Mark II containments.

Technical activities that support the development of a regulatory basis for the rulemaking include these components: (1) development of a core damage event tree and an accident progression event tree as front-end Probabilistic Risk Assessment (PRA) to identify and select risk-dominant accident sequences, (2) accident progression and source term analyses of selected accident sequences using MELCOR, (3) consequence analysis, based on MELCOR source terms using MACCS, and (4) risk assessment based on MACCS consequence results and PRA. The results verified that both water addition and venting are required to maintain containment structural integrity and that water addition is a beneficial strategy to influence containment conditions, cool core debris, and reduce radiological releases.

The staff is preparing an information paper to provide the draft regulatory basis to the Commission for consideration. The staff expects to provide that paper to the Commission in April 2015.

Mitigation of Beyond-Design-Basis Events Rulemaking

The NRC staff has also undertaken a consolidated rulemaking to address station blackout mitigation strategies and onsite emergency response capabilities. The scope of that rulemaking includes the following:

- All the requirements that were previously part of the Station Blackout Mitigation Strategies rulemaking, directed by COMSECY-13-0002, “Consolidation of Japan Lessons Learned Near-Term Task Force Recommendations 4 and 7 Regulatory Activities.”

- All the requirements that were to be part of the Onsite Emergency Response Capabilities rulemaking. This rulemaking, which stemmed from Recommendation 8 of the NTTF report, was directed by SRM-SECY-11-0137, “Prioritization of Recommended Actions to Be Taken in Response to Fukushima Dai-ichi Lessons Learned.” This portion of the consolidated rulemaking would also address NTTF Recommendation 10.2 concerning command and control and the qualifications of decision-makers. Command and control is also being addressed in industry guidance through the NEI 14-01, “Emergency Response Procedures and Guidelines for Extreme Events and Severe Accidents,” Revision 0.

- Numerous emergency preparedness actions are part of this rulemaking. They are also being implemented in conjunction with the implementation of EA-12-049 and through the development of guidance supporting the Recommendation 8 portion of the consolidated rulemaking. Those issues include:

  - Staffing and communication issues stemming from NTTF Recommendation 9.3, and also discussed in NTTF Recommendations 9.1 and 9.2. Staffing and communications are being addressed through Order EA-12-049 implementation guidance and NEI 12-01, which is referenced in NEI 12-06 and endorsed by the NRC in Interim Staff Guidance JLD-ISG-12-01.

  - Facilities and equipment issues stemming from NTTF Recommendation 9.3 and discussed in NTTF Recommendations 9.1 and 9.2. Facilities and equipment issues are being addressed through Order EA-12-049 implementation guidance and also NEI 13-06, "Enhancements to Emergency Response Capabilities for Beyond-Design-Basis Accidents and Events," Revision 0, which is under development.

  - Multiunit dose assessment issues stemming from NTTF Recommendation 9.3 and discussed in NTTF Recommendation 9.1. NEI 13-06 guidance will address multiunit dose assessment issues.

  - Training and exercise issues stemming from NTTF Recommendation 9.3 and discussed in NTTF Recommendations 9.1 and 9.2. Training and exercise issues are being addressed through EA-12-049 implementation guidance and also NEI 13-06 guidance.

  - Onsite emergency resources to support a station blackout affecting all units on a site including the need to deliver equipment to the site with offsite infrastructure degraded, stemming from NTTF Recommendation 11.1. Order EA-12-049 and supporting guidance is addressing this issue.

This consolidated rulemaking will address, either in requirements or through supporting implementation guidance, all of the recommendations in NTTF Recommendations 4, 7, 8, 9.1, 9.2, 9.3 (with the exception of maintenance of Emergency Response Data System (ERDS) capability throughout the accident), 10.2, and 11.1.

The proposed rule is due to the Commission by the end of April 2015.

<u>Onsite and Offsite Radiation Monitoring</u>

In addition to the radiation monitoring measures being put in place through Order EA-12-049 and associated guidance, and the Mitigation of Beyond-Design-Basis Events (MBDBE) rulemaking, as mentioned above, additional monitoring would be provided through the NRIA to the NRF. There is also a planned Tier 3 action to study the efficacy of real-time radiation monitoring onsite and within the emergency planning zones (EPZs), including consideration of AC independence and real-time availability on the Internet. The action also would call for training, in coordination with the appropriate Federal partners, on radiation, radiation safety, and the appropriate use of potassium iodide in the local community around each nuclear power plant. NRC staff plans to address these Tier 3 items in an advance notice of proposed rulemaking (ANPR) in fiscal year 2016 and will consider applicable NAS findings and recommendations in the process.

Hydrogen Control and Mitigation inside Containment or Other Buildings

In addition to the hydrogen control measures being put in place through Order EA-12-049 and associated guidance and the Mitigation of Beyond-Design-Basis Events Rulemaking, NTTF Recommendation 5.2 identified a need to reevaluate hardened vents for containment designs other than BWR Mark I and Mark II containments. NTTF Recommendation 6 is to identify insights from Fukushima Dai-ichi related to hydrogen control and mitigation inside containment or in other buildings and to determine whether more regulatory action is warranted. While these activities are separate, NRC staff expects that insights from carrying out the order related to severe-accident-capable vents for Mark I and Mark II containments, Order EA-13-109, will inform further evaluation and action for both activities.

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

During its review of the NTTF report, the Advisory Committee on Reactor Safeguards (ACRS) noted that the Fukushima Dai-ichi operators faced significant challenges in understanding the condition of the reactors, containments, and SFPs because the existing design-basis instrumentation was either lacking electrical power or providing erroneous readings. As a result, another recommendation was developed to address the regulatory basis for requiring reactor and containment instrumentation to be enhanced to withstand beyond-design-basis accident conditions. This was prioritized as a Tier 3 activity because it requires further staff study and depends on the outcome of other lessons-learned activities.

Once the NRC staff has accumulated sufficient knowledge and data, regulatory will be taken through the appropriate mechanism for any safety-significant instrumentation performance gaps.

Conclusion:

Staff concludes that ongoing or planned NRC and industry actions adequately address NAS Recommendation 5.1A.

Recommendation 5.1B:

As the nuclear industry and its regulator implement the actions referenced in Finding 5.1, they should give specific attention to improving resource availability and operator training to enable effective responses to beyond-design-basis events including, when necessary, developing and implementing ad hoc responses to deal with unanticipated complexities.

Attention to the following is specifically needed:

1. Staffing levels for emergencies involving multiple reactors at a site, that last for extended durations, and/or that involve stranded plant conditions.

2. Strengthening and better integrating emergency procedures, extensive damage mitigation guidelines, and severe accident management guidelines, in particular for:

    - Coping with the complete loss of AC and DC power for extended periods.

    - Depressurizing reactor pressure vessels and venting containments when DC power and installed plant air supplies (i.e., compressed air and gas) are unavailable.

    - Injecting low-pressure water when plant power is unavailable.

    - Transitioning between reactor pressure vessel depressurization and low-pressure water injection while maintaining sufficient water levels to protect the core from damage.

    - Preventing and mitigating the effects of large hydrogen explosions on cooling systems and containment.

    - Maintaining cold shut down in reactors that are undergoing maintenance outages when critical safety systems have been disabled.

3. Training of operators and plant emergency response organizations, in particular:

    - Specific training on the use of ad-hoc responses for bringing reactors to safe shutdown during extreme beyond-design-basis events.

    - More general training to reinforce understanding of nuclear plant system design and operation and enhance operators' capabilities for managing emergency situations.

The quality and completeness of the changes that result from this recommendation should be adequately peer reviewed.

Recommendation addressed by:

The NRC and the nuclear industry have taken actions to improve resource availability and operator training to enable effective responses to beyond-design-basis events, which include developing and carrying out ad hoc response protocols to deal with unanticipated event complexities.

Mitigation of Beyond-Design-Basis Events Rulemaking

In addition to the actions that have been taken in response to Order EA-12-049 and associated guidance, the NRC staff is developing the MBDBE rulemaking described in detail the response to Recommendation 5.1A. Elements of the rulemaking that specifically address Recommendation 5.1B include:

- Staffing levels for emergencies involving multiple reactors at a site, that last for extended durations or that involve stranded plant conditions;

- Strengthening and better integrating emergency procedures, the new guidelines developed in response to EA-12-049, extensive damage mitigation guidelines, and severe accident management guidelines; and

- Training of operators and plant emergency response organizations.

Peer Review of Industry Actions

In addition to NRC inspection activities, it is expected that industry's efforts to improve resource availability and operator training to enable effective responses to beyond-design-basis events will be peer reviewed independently by the Institute of Nuclear Power Operations (INPO).

The Kemeny Commission, which was established by President Jimmy Carter to investigate the March 1979 accident at the Three Mile Island nuclear power plant, recommended that the nuclear power industry establish a program that specifies appropriate safety standards, including those for management, quality assurance, and operating procedures and practices, and that conducts independent evaluations. In addressing those recommendations, the nuclear power industry established INPO, a not-for-profit organization headquartered in Atlanta, GA. INPO's mission "…is to promote the highest levels of safety and reliability—to promote excellence—in the operation of commercial nuclear power plants." To achieve its mission, INPO establishes performance objectives, criteria, and guidelines for the nuclear power industry, evaluates nuclear power plants regularly and in detail, and offers assistance to help nuclear power plants continually improve safety and performance.

INPO teams evaluate nuclear electric generating facilities by observing operations, analyzing processes, and reviewing performance data. With a focus on safety and reliability, these peer review teams assess the knowledge and performance of plant personnel, the condition of systems and equipment, the quality of programs and procedures, and the effectiveness of plant leadership. INPO teams are reviewing site implementation of post-Fukushima Dai-ichi changes, including training for plant operators and other emergency response personnel. It is expected that INPO will continue to provide periodic peer reviews in these areas as part of their ongoing plant evaluation activities.

Conclusion:

Staff concludes that ongoing or planned NRC and industry actions adequately address NAS Recommendation 5.1B.

Finding 5.2:

Beyond-design-basis events—particularly low-frequency, high-magnitude (i.e., extreme) events—can produce severe accidents at nuclear plants that damage reactor cores and stored spent fuel. Such accidents can result in the generation and combustion of hydrogen within the plant and release of radioactive material to the offsite environment. There is a need to better understand the safety risks that arise from such events and take appropriate countermeasures to reduce them.

Based on the above finding, the committee made the following recommendations:

Recommendation 5.2A:

The U.S. nuclear industry and the NRC should strengthen their capabilities for identifying, evaluating, and managing the risks from beyond-design-basis events. Particular attention is needed to improve the identification of such events; better account for plant system interactions and the performance of plant operators and other critical personnel in responding to such events; and better estimate the broad range of offsite health, environmental, economic, and social consequences that can result from such events.

Recommendation addressed by:

The NRC and the nuclear industry have taken actions to strengthen their capabilities for identifying, evaluating, and managing the risks from beyond-design-basis events.

Risks from Beyond-Design-Basis External Events

As learned from the Fukushima Dai-ichi accident, beyond-design-basis external events can result in the loss of all AC and DC power and loss of the ultimate heat sink. After the accident, the NRC initiated actions for U.S. nuclear power plants to reevaluate earthquake and flooding hazards at their sites using present-day information.

Seismic Hazard Reevaluations

The NRC issued a letter pursuant to 10 CFR 50.54(f) on March 12, 2012. That letter asked nuclear power plant licensees to use current regulations and guidance to reevaluate the seismic hazards that could affect their sites. If these newly reevaluated hazards are not bounded by the current design basis, the licensee will analyze them to determine whether interim measures for protection are needed. More evaluations are required if a site hazard exceeds established thresholds. These evaluations look at the systems and components that can be used to shut down a plant safely under the conditions of a station blackout and loss of normal access to the ultimate heat sink under conditions of a stronger seismic event, typically twice the design-basis safe shutdown earthquake. The NRC staff will review these evaluations to determine if additional regulatory action is necessary.

Flooding Hazard Reevaluations

The March 12, 2012, 10 CFR 50.54(f) letter also requested that all power reactor licensees reevaluate the flooding hazards that could affect their site. If the reevaluated flooding hazard at a site is not bounded by the current design-basis, licensees are asked to assess the plant's ability to cope with the reevaluated flood hazard (referred to as the integrated assessment). If the reevaluated hazard exceeds the capability of existing flood protection or mitigation, the 10 CFR 50.54(f) letter also requests licensees to describe interim actions taken or planned to address the reevaluated hazard. Examples of interim actions proposed by licensees include the use of sandbags or other temporary barriers, and use of FLEX strategies. The NRC staff will review the responses and determine whether regulatory actions are necessary to increase protection against flooding.

Consideration of Other Natural External Hazards

In a letter dated October 13, 2011, the ACRS recommended expanding NTTF Recommendation 2.1 to include natural external hazards other than seismic and flooding hazards. The Consolidated Appropriations Act, Public Law 112-074, directed the NRC to require reactor licensees to reevaluate the external hazards at their sites and to require updates to their design basis, if necessary.

Reevaluation of other natural external hazards was prioritized as a Tier 2 activity because of the lack of availability of the critical skill sets on the part of both the NRC staff and external stakeholders, as well as because the staff considered the seismic and flooding reevaluations to be of higher priority. The NRC intends to begin work on this recommendation as NRC and industry resources become available and as sufficient progress is made on the flooding and seismic reevaluations.

Performance of Plant Operators and Other Critical Personnel in Responding to Beyond-Design-Basis Events

Licensee actions being taken in response to Order EA-12-049 and associated guidance and actions which will be required by the MBDBE rulemaking, as described in detail the response to Recommendation 5.1A, adequately address Recommendation 5.2A. Those actions include:

- Staffing levels for emergencies involving multiple reactors at a site that last for extended durations or that involves stranded plant conditions;
- Strengthening and better integrating emergency procedures, the new guidelines developed in response to EA-12-049, extensive damage mitigation guidelines, and severe accident management guidelines; and
- Training of operators and plant emergency response organizations.

Offsite Consequences

The NRC and the Federal Emergency Management Agency (FEMA) are the primary Federal agencies responsible for radiological emergency preparedness in the U.S. The NRC is responsible for ensuring that U.S. nuclear plants are prepared for radiological emergencies and coordinates with FEMA, which oversees State and local agencies' preparedness for offsite actions. There is a Tier 3 action for NRC to work with FEMA, States, and other external stakeholders to evaluate insights from the Fukushima Dai-ichi accident to identify potential enhancements to the U.S. decision-making framework, including the concepts of recovery and reentry. Staff will consider applicable NAS findings and recommendations in this action.

State-of-the-Art Reactor Consequence Analyses (SOARCA) Project

The SOARCA report, which was published in November 2012, incorporates the results of more than 25 years of research to analyze the realistic outcomes of postulated severe reactor accidents, even though it is considered highly unlikely that such accidents could occur.

Accident phenomena and offsite consequences of severe reactor accidents have been the subjects of considerable research over the last several decades by the NRC. As a consequence of this research focus, analyses of severe accidents at nuclear power reactors are more detailed, integrated, and realistic than at any time in the past. A desire to leverage this capability to address conservative aspects of previous reactor accident analyses was a major motivating factor in the genesis of the SOARCA project. By applying modern analysis tools and techniques, the SOARCA project developed a body of knowledge regarding the realistic outcomes of select severe nuclear reactor accidents. To accomplish this objective, the SOARCA project's integrated modeling of accident progression and offsite consequences used both state-of-the-art computational analysis tools and best modeling practices drawn from the collective wisdom of the severe accident analysis community. The study focused on providing a realistic evaluation of accident progression, source term, and offsite consequences for select scenarios for the Peach Bottom Atomic Power Station and Surry Power Station. For both Peach Bottom and Surry, the SOARCA team modeled loss of all AC or station blackout scenarios caused by earthquakes more severe than anticipated in the plant's design, as this presented the most severe challenge to the plant operators, as well as offsite emergency responders, and had the highest probability of occurring.

By using the most current emergency preparedness practices and plant capabilities, as well as the best available modeling, these analyses are more realistic than past analyses. These analyses also consider mitigative measures (e.g., emergency operating procedures, severe accident management guidelines, and 10 CFR 50. 54(hh) requirements), contributing to a more realistic evaluation.

Key Findings:

- When operators are successful in using available onsite equipment during the accidents analyzed in SOARCA, they can prevent the reactor from melting, or delay or reduce releases of radioactive material to the environment;

- SOARCA analyses indicate that all modeled accident scenarios, even if operators are unsuccessful in stopping the accident, progress more slowly and release much smaller amounts of radioactive material than calculated in earlier studies;

- As a result, public health consequences from severe nuclear power plant accidents modeled in SOARCA are smaller than previously calculated;

- The delayed releases calculated provide more time for emergency response actions such as evacuating or sheltering for affected populations. For the scenarios analyzed, SOARCA shows that emergency response programs, if implemented as planned and practiced, reduce the risk of public health consequences;

- Both mitigated (operator actions are successful) and unmitigated (operator actions are unsuccessful) cases of all modeled severe accident scenarios in SOARCA cause essentially no risk of death during or shortly after the accident; and

- SOARCA's calculated longer-term cancer fatality risks for the accident scenarios analyzed are millions of times lower than the general U.S. cancer fatality risk;

The SOARCA project is designed to develop realistic estimates of the potential offsite effects on public health that might result from a nuclear power plant accident, in the event of very unlikely scenarios that could release radioactive material into the environment. Toward that end, this project is also designed to improve methods and models for realistically evaluating plant responses during severe accidents, including protective actions for the public (such as evacuation and sheltering) and the potential public health risk. The project's results will become the foundation for communicating aspects of severe accidents and updating information from older research studies.

Conclusion:

Ongoing or planned NRC and industry actions adequately address Recommendation 5.2A.

Recommendation 5.2B

The NRC should support industry's efforts to strengthen its capabilities by providing guidance on approaches and by overseeing independent review by technical peers (i.e., peer review).

Recommendation addressed by:

As was the case in the assessment of the potential risks of extreme seismic and flooding events and other lessons-learned initiatives, NRC staff will coordinate with external stakeholders to develop guidance to describe acceptable methods for finding, evaluating, and managing the risks from other extreme external events.

In addition to NRC inspection activities, since INPO teams are reviewing industry action taken in response to the accident, it is expected that INPO will continue to provide periodic peer reviews in these areas as part of their ongoing plant evaluation activities.

Conclusion:

Ongoing or planned NRC and industry actions adequately address Recommendation 5.2B.

Recommendation 5.2C:

As the U.S. nuclear industry and the NRC carry out the actions in Recommendation 5.2A, they should pay particular attention to the risks from beyond-design-basis events that have the potential to affect large geographic regions and multiple nuclear plants. These include earthquakes, tsunamis, and other geographically extensive floods, and geomagnetic disturbances.

Recommendation addressed by:

The NRC and nuclear industry are addressing beyond-design-basis events that have the potential to affect large geographic regions and multiple nuclear plants as part of ongoing or planned initiatives.

Potential Risks from Beyond-Design-Basis External Events

The response to Recommendation 5.2A above offers details concerning earthquake and flooding reassessments and NRC plans to assess other external hazards.

In addition to the actions taken in response to Order EA-12-049 and associated guidance and actions being taken as a result of the reassessment of seismic and flooding risks, the MBDBE rulemaking—described in detail the response to Recommendation 5.1A—will result in a more integrated accident response capability, including an enhanced emergency preparedness capability for beyond-design-basis affecting multiple units at the same site.

As the NRC staff evaluates licensees' strategies for complying with Order EA-12-049 and the rule, a review of the licensees' ability and preparations to carry out these strategies in the event of large-scale regional disturbances is evaluated as well.

Geomagnetic Disturbances

The NRC is considering geomagnetic disturbances in its rulemaking process due to the potential inability to resupply diesel fuel for emergency diesel generators in the event of a widespread, long-term loss of the electrical grid caused by a solar storm that could potentially affect multiple States and multiple reactor sites for a time period of weeks to months.

Specifically, this issue was discussed in a petition for rulemaking submitted by a member of the public. The petitioner asserts that the North American commercial electric power grids are vulnerable to prolonged outage caused by extreme space weather, such as coronal mass ejections and associated geomagnetic disturbances and therefore cannot be relied on to provide continual power for active cooling and/or water makeup of SFPs. Moreover, the petitioner asserts that existing means for providing onsite backup power are designed to operate for only a few days, while spent fuel requires active cooling for several years after removal of the fuel rods from the reactor core.

The staff's rationale for considering this issue as part of the NRC's rulemaking process is explained in a *Federal Register* notice (77 FR 74788; December 18, 2012) responding to a petition for rulemaking.

> However, in the event of a widespread electrical transmission system blackout for an extended duration (beyond 7 days and up to several months), it may not be possible to transport these and other necessary offsite resources to the affected NPPs [nuclear power plants] in a timely manner. Thus, government assistance (local, State, or Federal) may be necessary to maintain the capability to safely shutdown nuclear plants and cool spent fuel pools in the affected areas. Prior planning is needed to efficiently and effectively use government resources to ensure protection of public health and safety. Current NRC regulations do not require power reactor licensees to undertake mitigating efforts for prolonged grid failure scenarios that could be caused by geo-magnetically-induced currents (GICs) resulting from an extreme solar storm.

The NRC will review and analyze the underlying technical and policy issues relevant to the petition as part of its normal petition for rulemaking process.

In November 2014, in his role as Chair of the National Science and Technology Council, and on behalf of the President of the U.S., Dr. John Holdren chartered the interagency Space Weather Operations, Research, and Mitigation Task Force. The Task Force, in which the NRC is participating, will develop a National Space Weather Strategy that will articulate high-level strategic goals for enhancing our Nation's preparedness for a severe space weather event. In addition, a space weather action plan will be developed that will establish a process to carry out the National strategy. Both the strategy and the action plan are expected to be completed in 2015.

Conclusion:

Ongoing or planned NRC and industry actions adequately address Recommendation 5.2C.

Finding 5.3:

Four decades of analysis and operating experience have demonstrated that nuclear plant core-damage risks are dominated by beyond-design-basis accidents. Such accidents can arise, for example, from multiple human and equipment failures, violations of operational protocols, and extreme external events. Current approaches for regulating nuclear plant safety, which have been traditionally based on deterministic concepts such as the design-basis accident, are clearly inadequate for preventing core-melt accidents and mitigating their consequences. Modern risk assessment principles are beginning to be applied in nuclear reactor licensing and regulation. The more complete application of these principles in licensing and regulation could help to further reduce core melt risks and their consequences and enhance the overall safety of all nuclear plants, especially currently operating plants.

Based on the above finding the committee made the following recommendation:

Recommendation 5.3:

The NRC should further incorporate modern risk concepts into its nuclear reactor safety regulations. This effort should utilize the strengthened capabilities for identifying and evaluating risks that were described in Recommendation 5.2A.

Recommendation addressed by:

The NRC is in the process of considering a number of actions to improve usage of risk insights in the NRC's regulatory processes.

The Office of Nuclear Regulatory Research currently has a full-scope site Level 3 PRA project underway and plans to apply the results to NRC's regulatory framework.

The NRC is exploring initiatives that could expand using PRA in NRC's regulatory framework. For example, the staff is developing a notation vote paper that would offer the Commission options for implementing a risk management regulatory framework. The staff intends to provide the paper to the Commission by December 2015.

In addition to exploring a risk-management regulatory framework, the staff is preparing options for a risk-prioritization initiative. Such an initiative, if approved by the Commission, would allow licensees to use risk information to request plant-specific schedules for carrying out regulatory activities. Such an initiative could encourage using PRA further. The staff will present these options to the Commission in a notation vote paper due in March 2015.

The NRC has also formed the Risk-Informed Steering Committee to provide strategic direction and guidance to the NRC staff for the purpose of resolving a number of policy and technical issues in order to expand the use of PRA as a tool for regulatory decision-making.

A summary of the NRC's risk-informed activities can be found at the following site http://www.nrc.gov/about-nrc/regulatory/risk-informed/rpp.html. These and other initiatives are being used to ensure that the NRC's oversight programs realize the benefits of modern risk-assessment principles.

Conclusion:

Ongoing or planned NRC and industry actions adequately address Recommendation 5.3.

Finding 6.1:

The Fukushima Dai-ichi accident revealed vulnerabilities in Japan's offsite emergency management. The competing demands of the earthquake and tsunami diminished the available response capacity for the accident. Implementation of existing nuclear emergency plans was overwhelmed by the extreme natural events that affected large regions, producing widespread disruption of communications, electrical power, and other critical infrastructure over an extended period of time. Additionally:

1. Emergency management plans in Japan at the time of the Fukushima Dai-ichi accident were inadequate to deal with the magnitude of the accident requiring emergency responders to improvise.

2. Decision-making processes by government and industry officials were challenged by the lack of reliable, real-time information on the status of the plant, offsite releases, accident progression, and projected doses to nearby populations.

3. Coordination among the central and local governments was hampered by limited and poor communications.

4. Protective actions were improvised and uncoordinated, particularly when evacuating vulnerable populations (e.g., the elderly and sick) and providing potassium iodide.

5. Different and revised radiation standards and changes in decontamination criteria and policies added to the public's confusion and distrust of the Japanese government.

6. Cleanup of contaminated areas and possible resettlement of populations are ongoing efforts three years after the accident with uncertain completion timelines and outcomes.

7. Failure to prepare and implement an effective strategy for communication during the emergency contributed to the erosion of trust among the public for Japan's government, regulatory agencies, and the nuclear industry.

Finding 6.2:

The committee did not have the time or resources to perform an in-depth examination of U.S. preparedness for severe nuclear accidents. Nevertheless, the accident raises the question of whether a severe nuclear accident such as occurred at the Fukushima Dai-ichi plant would challenge U.S. emergency response capabilities because of its severity, duration, and association with a regional-scale natural disaster. The natural disaster damaged critical infrastructure and diverted emergency response resources.

Based on the above findings, the committee made the following recommendations:

Recommendation 6.2A:

The nuclear industry and organizations with emergency management responsibilities in the U.S. should assess their preparedness for severe nuclear accidents associated with offsite regional-scale disasters. Emergency response plans, including plans for communicating with affected populations, should be revised or supplemented as necessary to ensure that there are scalable and effective strategies, well-trained personnel, and adequate resources for responding to long-duration accident/disaster scenarios involving:

1. Widespread loss of offsite electrical power and severe damage to other critical offsite infrastructure, for example communications, transportation, and emergency response infrastructure.

2. Lack of real-time information about conditions at nuclear plants, particularly with respect to releases of radioactive material from reactors and/or spent fuel pools.

3. Dispersion of radioactive materials beyond the 10-mile emergency planning zones for nuclear plants that could result in doses exceeding one or more of the protective action guidelines.

Recommendation addressed:

The Fukushima Dai-ichi accident highlighted the complexities of emergency response during a nuclear accident coupled with an offsite disaster and resultant long-term loss of off-site electrical power. The NRC noted several emergency preparedness and response enhancements in light of the accident.

Staff comments in response to Recommendation 5.1A address Recommendation 6.2. Elements of the rulemaking that pertain specifically to this recommendation include:

- Staffing and communications;
- Facilities and equipment;
- Multiunit dose assessment;

- Training and exercises; and

- Onsite emergency resources to support multiunit with station blackout, including the need to deliver equipment to the site with offsite infrastructure degraded.

Licensee's emergency plans must describe the means for determining the magnitude of, and for continually assessing the effect of, the release of radioactive materials, including from all reactor core and spent fuel pool sources at a site. In addition, licensees must describe the emergency action levels that will classify the event and trigger protective actions, if necessary, and notify appropriate offsite officials. These emergency action levels are to be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. Licensees also must have equipment at the site for personnel monitoring and equipment for determining the magnitude of, and for continuously assessing the effect of, the release of radioactive materials, including from all reactor core and spent fuel pool sources.

Information Notice 2007-12 alerted licensees to reconsider the systems and equipment used to communicate with first responders in light of lessons learned from the terrorist attacks of 9/11 and the effect of Hurricane Katrina on offsite communications links. In the March 12, 2012, 10 CFR 50.54(f) letter, licensees were requested to assess their current communications systems and equipment used during an emergency event, identify any enhancements that may be needed, and assess the means to power the new and existing communications equipment onsite and offsite during a prolonged station blackout event.

The NRC has always considered as part of its emergency planning regulations and guidance that protective actions could be necessary beyond the 10-mile emergency planning zone (EPZ). For example, NUREG-0654/FEMA-REP-1 (1980), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," states, in part:

> In particular emergency, protective actions might well be restricted to a small part of the planning zone. On the other hand, for the worst possible accidents, protective actions would need to be taken outside the planning zones. The current EPZs provide for a comprehensive emergency planning framework that would allow expansion of the response efforts beyond the designated distances should events warrant such an expansion. The 10-mile emergency planning basis establishes an infrastructure similar to that used by other offsite response organizations, such as police and fire departments. The infrastructure consists of emergency organizations, communications capabilities, training, and equipment that can be used in the event of an accident at a facility. Coordination is enhanced by the practice of having offsite response organizations, which include local, State, and Federal responders, participate in training exercises with the licensee.

The Commission affirmed the staff's position on the adequacy of the current emergency planning zone sizes in SRM-SECY-13-0135, "Denial of Petition for Rulemaking Requesting Amendments Regarding Emergency Planning Zone Size."

Conclusion:

Ongoing or planned NRC and industry actions adequately address Recommendation 6.2A.

Recommendation 6.2B:

The nuclear industry and organizations with emergency management responsibilities in the U.S. should assess the balance of protective actions (e.g., sheltering-in-place, evacuation, relocation, and distribution of potassium iodide) for offsite populations affected by severe nuclear accidents and revise the guidelines as appropriate.

Particular attention should be given to the following issues:

1. Protective actions for special populations (children, ill, elderly) and their caregivers.

2. Long-term impacts of sheltering-in-place, evacuation and/or relocation, including social, psychological, and economic impacts.

3. Decision-making for resettlement of evacuated populations in areas contaminated by radioactive material releases from nuclear plant accidents.

Recommendation addressed by:

The NRC is responsible for determining the overall state of emergency preparedness at nuclear power plants. The NRC relies on FEMA for assessing offsite radiological emergency response plans and preparedness, and determining the adequacy and capability of putting into place the offsite plans. The NRC reviews FEMA's findings and determinations in conjunction with the NRC's onsite findings for the purpose of making a determination of reasonable assurance for protection of public health and safety.

There is a Tier 3 action for NRC to work with FEMA, States, and other external stakeholders to evaluate insights from the Fukushima Dai-ichi accident to identify potential enhancements to the U.S. decision-making framework, including the concepts of recovery and reentry. NRC staff plans to address this item in an ANPR, a tool that allows the staff to solicit early written stakeholder input on a new potential rulemaking effort. The NRC staff expects to issue the ANPR in fiscal year 2016.

The joint NRC-FEMA guidance document NUREG-0654/FEMA REP-1 included the need to consider special populations in developing protective actions. However, the definition of such populations was defined narrowly. Lessons learned from Hurricane Katrina in 2005 found the need to expand the definition of special populations to include those: (1) who have disabilities but are not institutionalized, (2) who are elderly and living independently, (3) who are "latch key" children, and (4) those that are from diverse cultures. Radiological emergency planning in NUREG-0654/FEMA-REP-1 included requirements to consider those who have limited English proficiency, don't speak English, or lack transportation.

Evacuation of local populations is not specifically a nuclear power plant issue; and FEMA addresses evacuation of all populations, including those with special needs, within its Comprehensive Preparedness Guide (CPG) 101. As discussed in CPG 101, "Even though each hazard's characteristics (e.g., speed of onset, size of the affected area) are different, the general tasks for conducting an evacuation and shelter operations are the same." CPG 301, "Emergency Management Planning Guide for Special Needs Populations," was developed to help tribal, state, territorial, and local governments develop emergency plans for people with functional needs. CPG 302, “Incorporating Household Pets and Service Animals Considerations Into Emergency Operations Plans: A Guide for State, Territorial, Tribal, and Local Governments,” is based on lessons learned from Hurricane Katrina and offers planning considerations for a variety of hazards, security, and emergency functions including nuclear power plant accidents.

According to the National Preparedness Goals, the recovery process includes the need to restore and improve health and social services networks, promoting the resilience, independence, health (including behavioral health), and well-being of the whole community. The Federal agencies tasked by the NRF to respond to disasters (including nuclear power plant accidents) must consider long-term social, psychological and economic effects.

The U.S. Environmental Protection Agency (EPA) is responsible for radiation protection standards for the general population, which includes dose reduction through offsite remediation of nuclear or radiological incidents. The NRC is part of the interagency committee that has closely supported EPA in its revision to its Protective Action Guidelines Manual. The updated Protective Action Guides and Planning Guidance for Nuclear Incidents, draft published in April 2013 for interim use, addresses long-term resettlement and recovery in Chapter 3, “Intermediate Phase Protective Action Guides,” and Chapter 4, “Guidance for the Late Phase.” The management and staff of the H. B. Robinson Nuclear Plant, along with the NRC, FEMA, and other Federal agencies, will participate in a full-scale exercise in July 2015 to test the intermediate and long-term response to a nuclear power plant accident accompanied by a hypothetical release of radioactive materials.

Conclusion:

Ongoing or planned NRC and industry actions adequately address Recommendation 6.2B.

Finding 7.1:

While the Government of Japan acknowledged the need for a strong nuclear safety culture prior to the Fukushima Dai-ichi accident, TEPCO and its nuclear regulators were deficient in establishing, implementing, and maintaining such a culture. Examinations of the Japanese nuclear regulatory system following the Fukushima Dai-ichi accident concluded that regulatory agencies were not independent and were subject to regulatory capture.

Finding 7.2:

The establishment, implementation, maintenance, and communication of a nuclear safety culture in the U.S. are priorities for the U.S. nuclear power industry and the NRC. The U.S. nuclear industry, acting through INPO, has voluntarily established nuclear safety culture programs and mechanisms for evaluating their implementation at nuclear plants. The NRC has published a policy statement on nuclear safety culture, but that statement does not contain implementation steps or specific requirements for industry adoption.

Based on the above findings, the committee made the following recommendations:

Recommendation 7.2A:

The NRC and the U.S. nuclear power industry must maintain and continuously monitor a strong nuclear safety culture in all of their safety-related activities. Additionally, the leadership of the NRC must maintain the independence of the regulator. The agency must ensure that outside influences do not compromise its nuclear safety culture and/or hinder its discussions with and disclosures to the public about safety-related matters.

Recommendation addressed by:

The NRC has a long history of recognizing the importance of a strong safety culture, which includes maintaining the independence of the regulator.

Safety Culture

The NRC has long known the importance of a strong nuclear safety culture. In response to a 1989 incident at the Peach Bottom Nuclear Power Plant, the NRC issued a "Policy Statement on the Conduct of Nuclear Power Plant Operations," which described the Commission's expectation that licensees place appropriate emphasis on safety in the operation of nuclear power plants. That policy statement placed an emphasis on the personal dedication and accountability of all individuals engaged in any activity that has a bearing on the safety of nuclear power plants. Additionally, the policy statement underscored management's responsibility for fostering a healthy safety culture at each facility and for providing a professional working environment in the control room—and throughout the facility—to ensure safe operations.

In 1996, after an incident at the Millstone Nuclear Power Station in which workers faced retaliation for whistleblowing, the Commission issued another policy statement, "Freedom of Employees in the Nuclear Industry to Raise Safety Concerns without Fear of Retaliation." This policy statement described the NRC's expectation that all licensees establish a safety-conscious work environment (SCWE), in which workers feel free to raise nuclear safety concerns without fear of harassment, intimidation, retaliation, or discrimination. An SCWE is an important attribute of a strong nuclear safety culture.

In 2002, investigations into the discovery of degradation of the reactor pressure vessel head at Davis-Besse Nuclear Power Station revealed that safety culture weaknesses were a root cause of the event. The NRC took significant steps within the Reactor Oversight Process (ROP) to strengthen the agency's ability to monitor licensee performance effectively and detect potential safety culture weaknesses during inspections and performance assessments. Regulatory Issue Summary 2006-13, "Information on the Changes Made to the Reactor Oversight Process to More Fully Address Safety Culture," was issued on July 31, 2006, to supply information to reactor licensees on the revised ROP. Most notably, the NRC revised the existing cross-cutting areas of human performance, problem identification and resolution, and SCWE to incorporate aspects important to safety culture. The intent of the ROP revisions was threefold:

- To provide better opportunities for the NRC staff to consider safety culture weaknesses and to encourage licensees to take appropriate actions before significant performance degradation occurs;

- To provide the NRC staff a process to determine the need to specifically evaluate a licensee's safety culture after performance problems has resulted in placing the licensee in the degraded cornerstone column of the action matrix; and

- To provide the NRC staff a structured process to evaluate the licensee's safety culture assessment and to assess a safety culture independently for a licensee in the multiple/repetitive degraded cornerstone column of the action matrix.

In 2004, also in response to events at Davis-Besse Nuclear Power Station, INPO published a document titled, "Principles for a Strong Nuclear Safety Culture," which described principles and attributes of a healthy nuclear safety culture as developed by an industry advisory group. In 2009, in partnership with NEI and INPO, the nuclear power industry began an effort to enhance safety culture. The industry's process for monitoring and improving safety culture used INPO's principles and attributes of a healthy nuclear safety culture as a framework and was described in the document NEI 09-07, "Fostering a Strong Nuclear Safety Culture."

In 2008, at the Commission's direction, the NRC staff began an effort to expand the safety culture policy to address the unique aspects of security and ensure applicability to all licensees and certificate holders. The NRC engaged in a unique collaborative effort with stakeholders to develop a definition of nuclear safety culture and a list of traits that describe that safety culture. Since Agreement States have entered into agreements with the NRC that give them the authority to license and inspect byproduct, source, or special nuclear materials used or possessed within their borders, they were included in this effort. The goal of this effort was to develop a model that could be applied to any of the diverse stakeholders responsible for the safe and secure use of nuclear materials.

The final NRC Safety Culture Policy Statement (SCPS) was published on June 14, 2011. The SCPS provides the NRC's expectation that individuals and organizations performing regulated activities establish and maintain a healthy safety culture that recognizes the safety and security significance of their activities and the nature and complexity of their organizations and functions. Because safety and security are the primary pillars of the NRC's regulatory mission, consideration of both safety and security issues, commensurate with their significance, is an underlying principle of the SCPS.

NRC Independence

In achieving the agency's mission, the NRC adheres to the principles of good regulation: independence, openness, efficiency, clarity, and reliability. The agency puts these principles into practice with effective, realistic, and timely regulatory actions, consistent with our organizational values and our open, collaborative work environment.

The Energy Reorganization Act, signed into law October 11, 1974, established the NRC. Under the Atomic Energy Act of 1954, a single agency, the Atomic Energy Commission, had responsibility for developing and producing nuclear weapons, and for developing and regulating civilian uses of nuclear materials.

The Act of 1974 split these functions, assigning to one agency, now the Department of Energy (DOE), the responsibility for the development and production of nuclear weapons, promotion of nuclear power, and other energy-related work, and assigning to the NRC the responsibility for the oversight of civilian use of nuclear materials (which does not include regulation of defense nuclear facilities). The Act of 1974 gave the Commission its collegial structure and established its major offices. A later amendment to the Act also offered protections for employees who raise nuclear safety concerns.

The Senate and House Committees with jurisdiction over domestic nuclear regulatory activities are the Senate Committee on Environment and Public Works and the House Committee on Energy and Commerce. Within those Committees, the subcommittees responsible for NRC legislation and oversight are the Senate Subcommittee on Clean Air and Nuclear Safety, and the House Subcommittee on Energy and the Environment. These two subcommittees have jurisdiction over authorizing legislation for the NRC and constitute the principal Congressional oversight of the NRC.

It should also be noted NRC receives independent oversight from the NRC's Office of the Inspector General and the U.S. Government Accountability Office.

The U.S. is a contracting party to the Convention on Nuclear Safety (CNS), which states, in part: "Each Contracting Party shall take the appropriate steps to ensure an effective separation between the functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy." In this regard, every contracting party provides a national report and submits it for peer review every three years. These reports describe how each party fulfills its obligations under the CNS, including maintaining the independence of the regulatory body. The NRC prepares the U.S. national report and leads the U.S. delegation to the review meetings. Thus, the leadership of the NRC maintains an ongoing focus on sustaining its independence and submits to review by international regulators in that regard.

Conclusion:

Recommendation 7.2A. is adequately addressed by ongoing activities.

Recommendation 7.2B:

The U.S. nuclear industry and the NRC should examine opportunities to increase the transparency of, and communication about, their efforts to assess and improve their nuclear safety cultures.

Recommendation addressed by:

The NRC maintains a public safety culture website at http://www.nrc.gov/about-nrc/safety-culture.html. The website allows the public and other stakeholders to access outreach materials that can be used to educate stakeholders about safety culture and the NRC's SCPS.

In March 2014, the staff published NUREG-2165, "Safety Culture Common Language," which documents the outcomes of public workshops to develop a common language to describe safety culture in the nuclear industry. The purpose of this initiative was to align NRC and licensee terminology when describing safety culture at nuclear power facilities. These workshops, held in December 2011, April 2012, November 2012, and January 2013, included subject matter experts from the NRC, the nuclear industry, and the public. The common language was completed and agreed upon at the January 2013 workshop. The NRC staff uses the agreed-upon common language to carry out elements of its programs that oversee regulated activities. Parts of the common language were incorporated into the ROP for operating nuclear reactors. All changes to oversight programs, including the ROP, have been documented in their associated inspection manual chapters and inspection procedures.

Conclusion:

Recommendation 7.2B is adequately addressed by ongoing activities.

Overall Conclusion

Overall, the staff concludes that ongoing or planned NRC and industry activities address NAS's recommendations. The staff will consider NAS's insights, along with other studies and information from the Fukushima Dai-ichi accident, in future actions.

On March 10, 2015, NRC staff met with Mr. Kevin D. Crowley, NAS Study Director for the Fukushima Dai-ichi report to discuss staff's review, findings, and conclusions.