POLICY ISSUE
(Information)

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FOR: The Commissioners

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SUBJECT: STAFF EVALUATION OF APPLICABILITY OF LESSONS LEARNED FROM THE FUKUSHIMA DAI-ICHI ACCIDENT TO FACILITIES OTHER THAN OPERATING POWER REACTORS

PURPOSE:

The purpose of this paper is to inform the Commission of the results of the staff's evaluation of the applicability of lessons learned from the accident at the Fukushima Dai-ichi nuclear power plant in Japan to facilities other than operating power reactors. This paper does not address any new commitments or resource implications.

SUMMARY:

Shortly after the March 11, 2011, earthquake and tsunami that led to the accident at the Fukushima Dai-ichi nuclear power plant in Japan, the U.S. Nuclear Regulatory Commission (NRC) staff was directed to undertake near-term and longer-term actions to assess the accident and identify lessons learned for U.S. nuclear facilities ("Tasking Memorandum – COMGBJ-11-0002 – NRC Actions Following the Events in Japan," dated March 23, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110820875)). The Near-Term Task Force (NTTF) was created to review the events and the possible implications for the safety of U.S. nuclear power plants. On July 12, 2011, the NTTF issued its report, titled "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan" (ADAMS Accession No. ML11186A950), which includes 12 recommendations. NRC staff prioritized the NTTF recommendations and has been actively implementing these recommendations through orders, requests for information, rulemaking, and other regulatory initiatives.

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As one of the longer-term activities, COMGBJ-11-0002 also directed the staff to assess the applicability of the lessons learned from the accident to non-operating reactors and non-reactor facilities. Very shortly after the accident, NRC staff performed limited assessments to ensure that no immediate safety concerns existed at these facilities and took safety measures where necessary, such as an independent verification of fuel facility licensees' abilities to prevent or mitigate the consequences of events that could challenge the safety or licensing bases of those facilities. With insights gained from NRC activities related to power reactors and from the results of inspections at fuel cycle facilities, NRC staff has more fully evaluated issues and possible actions related to other NRC-licensed materials, devices, and facilities. NRC staff's detailed evaluation can be found in Enclosure 1 of this paper. Based on the evaluation of lessons learned from the Fukushima accident, NRC staff concludes that with the exception of some limited scope additional analyses, there is no need for regulatory action for facilities other than operating power reactors in light of the accident.

NRC staff is continuing additional assessment related to Fukushima lessons learned for NRC-regulated fuel facilities and a small subset of research and test reactors (RTRs). For all other facilities and licensees reviewed, NRC staff has determined that further assessments are not needed based on Fukushima lessons learned and that the existing regulatory requirements and processes ensure adequate protection of public health and safety.

BACKGROUND:

The series of events that resulted in core damage at the Fukushima Dai-ichi nuclear power plant was initiated by an earthquake and the resulting tsunami. The tsunami exceeded the design bases for the facility and inundated emergency power systems, leading to an extended loss of electrical power and, eventually, core damage and release of large amounts of radioactive material.

In response to the Fukushima Dai-ichi accident, the Commission established the NTTF to conduct a systematic and methodical review of NRC processes and regulations and determine if the agency should make improvements to its regulatory program. COMGBJ-11-0002 stated that "…applicability of the lessons learned to non-operating reactor and non-reactor facilities should also be explored." In SECY-11-0117, "Proposed Charter for the Longer-Term Review of Lessons Learned from the March 11, 2011, Japanese Earthquake and Tsunami," dated August 26, 2011 (ADAMS Accession No. ML11231A886), NRC staff defined the scope as follows:

> The scope of the NRC's longer-term review will include those items identified in the Chairman's tasking memorandum for longer-term review, recommendations for evaluation that were provided by the Near-Term Task Force and are approved by the Commission, and any other review topics the Commission directs. The scope of the steering committee's review will include power and non-power reactors, non-operating reactors, and non-reactor NRC licensees, and will be informed by interactions with external stakeholders.

In SECY-11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011 (ADAMS Accession No. ML11272A111), NRC staff developed longer-term plans to address the recommendations from the NTTF and other stakeholders. In SECY-11-0137, the staff indicated that it would continue to evaluate the applicability of lessons learned to licensed facilities other than power reactors (e.g., RTRs, independent spent fuel storage installations, and reactors that have permanently ceased operations but still maintain fuel in a spent fuel pool), and take appropriate actions.

For fuel cycle facilities, the staff issued Temporary Instruction (TI) 2600/15, "Evaluation of Licensee Strategies for the Prevention and/or Mitigation of Emergencies at Fuel Facilities," dated September 30, 2011 (ADAMS Accession No. ML111030453). The inspection activities of TI 2600/15 included those licensees required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 70, "Domestic Licensing of Special Nuclear Material," to conduct an integrated safety analysis (ISA); those licensees under 10 CFR Part 40, "Domestic Licensing of Source Material," required by their license to conduct an ISA; and currently operating certificate holders under 10 CFR Part 76, "Certification of Gaseous Diffusion Plants." As described in TI 2600/15, the types of events NRC assessed included seismic hazards, external flooding hazards, internal flooding hazards, wind and tornado loading, extended loss of alternating current or emergency power, and fires.

NRC staff evaluated the findings from TI 2600/15 using its normal inspection processes. As a result of the inspection, NRC staff identified a concern at the Honeywell Metropolis Works Facility. This facility converts uranium concentrate received from uranium mines to uranium hexafluoride. Specifically, as a result of its inspection activities, the staff found that the facility may not have been adequately protected from an unlikely but credible event, such as an earthquake or tornado. After the inspection, the licensee agreed to fortify its building and processing equipment. The licensee shut down its facility until NRC staff confirmed that appropriate upgrades had been made. In August 2013, NRC authorized the licensee to restart the facility following the completion of the necessary upgrades.

With insights gained from NRC activities related to power reactors, such as seismic and flooding walkdowns, and the results of inspections at fuel cycle facilities, NRC staff has fully evaluated the issues and possible actions related to NRC licensees other than operating power reactors. Similar to TI 2600/15, the types of events that NRC staff assessed for these facilities included postulated external events, seismic hazards, external flooding hazards, internal flooding hazards, wind and tornado loading, extended loss of alternating current or emergency power, and fires, to determine if existing regulatory requirements appropriately address such hazards. In addition to the evaluation of initiating events and external hazards, NRC staff assessed these licensees qualitatively in terms of (1) policy issues related to Fukushima, (2) the NTTF's findings and recommendations, and (3) other domestic and international studies and evaluations. NRC staff's review was broad in scope and was not limited to specific recommendations and considerations provided by the NTTF, which tend to be discussed in the context of nuclear power reactors.

DISCUSSION:

NRC staff's assessment includes a screening and evaluation of specific types of NRC licensees other than operating power reactors. These evaluations were completed through document reviews of license requirements and the regulatory framework, and internal meetings with subject matter experts. The extent of these evaluations depended on the nature of the licensed facility activities, devices, or material (e.g., amount of radioactive material, complexity in a given facility's design, inherent safety, etc.).

NRC staff factored several other evaluations and actions into its reviews. For example, as discussed above, fuel cycle facilities were inspected shortly after the Fukushima Dai-ichi accident and appropriate actions were taken to ensure adequate protection of public health and safety (including the multi-month shutdown of the Honeywell Metropolis Works Facility to allow for upgrades to the facility). Another example of such an evaluation is the NRC staff's assessment of the safety of RTRs in Pacific coastal States in light of the tsunami forecasted to impact the Pacific coast of the United States as a result of the March 11, 2011, earthquake off the coast of Japan. As part of that evaluation, NRC staff concluded that none of the five RTRs in the Pacific States were vulnerable to a tsunami reaching the locations of the facilities. In cases where NRC staff performed previous evaluations, they are discussed in the facility-specific assessments supplied in Enclosure 1 of this paper.

NRC staff's assessments address the following general areas:

- spent fuel storage and transportation
- fuel facilities
- radioactive material users
- irradiators licensed in accordance with 10 CFR Part 36, "Licenses and radiation safety requirements for irradiators"
- low-level waste disposal facilities
- uranium recovery facilities and uranium mill tailings
- decommissioning reactors and complex materials facilities
- non-power reactors (e.g., RTRs)

For each of these areas NRC staff assessment provides the following:

- the current regulatory framework for the area
- relevant pre-Fukushima assessments
- insights from experience gained from international activities, if applicable
- an assessment of the facilities' capabilities to address or mitigate the following events: flood, seismic, high wind and missiles, lightening, snow and ice loads, temperature extremes, fire, and loss of power
- an assessment of each of the 12 NTTF recommendations for each of the areas

Results of Stakeholder Interactions:

NRC staff provided a draft white paper to stakeholders for their review and comment, which contained much of the staff's assessment found in Enclosure 1 of this paper. As appropriate, changes have been made to the NRC staff's assessment found in Enclosure 1 as a result of comments received from these stakeholders. The stakeholder comments were gathered in two ways:

- For classes of licensees that are regulated by both Agreement States and the NRC, NRC staff solicited comments on the NRC staff's draft white paper through a letter to Agreement States dated February 20, 2015 (ADAMS Accession No. ML15050A430 (nonpublic)). The February 20, 2015, letter included the sections of the NRC staff's draft white paper relevant to the Agreement States. This included sections of the NRC staff's draft white paper associated with: (1) radioactive material users, (2) irradiators licensed in accordance with 10 CFR Part 36, (3) low-level waste disposal facilities, (4) uranium recovery and uranium mill tailings, and (5) decommissioned reactors and complex materials facilities.

- For licensees and other external stakeholders, the staff solicited comments through a March 13, 2015, Category 3 public meeting[1] on the topic. The NRC staff's draft white paper was supplied as part of the notice for the public meeting and stakeholders had the opportunity to offer written comments through an email address associated with the meeting or by verbal comments at the meeting. The NRC staff's draft white paper

[1] Category 3 public meetings are fully engaged discussions between the NRC and the public (as well as stakeholders that might include other government agencies, the industry and others). Public participation is actively sought at this type of meeting, which has the widest participation opportunities and is specifically tailored for the public to comment or ask questions.

provided to support the March 13, 2015, public meeting included the same material provided in the February 20, 2015, letter to the Agreement States, plus the following three additional draft assessments: (1) spent fuel storage and transportation, (2) fuel facilities, and (3) NRC-licensed non-power reactors.

Agreement States' Comments

The Commonwealth of Virginia and the Organization of Agreement States (OAS) provided comments on the NRC staff's draft white paper in letters dated March 19, 2015 (ADAMS Accession No. ML15083A303), and March 23, 2015 (ADAMS Accession No. ML15083A281), respectively. The two letters generally provided the same comments. The major comment was that the Commonwealth of Virginia and the OAS Board agreed with NRC staff's conclusion that no further study or regulatory action is warranted beyond those already being implemented as part of the regulatory oversight of the licensees discussed in the February 20, 2015, draft white paper. Other comments provided in the two letters included proposed technical clarifications and requests for more explicit statements recognizing the role that the Agreement States have in regulating some of the facilities that are within the scope of the NRC staff's assessment.

NRC staff generally incorporated the comments provided in the two letters and adjusted the assessment found in Enclosure 1 of this document. A more detailed discussion of the NRC staff's resolution of the Agreement State comments can be found in an NRC letter to OAS dated April 3, 2015 (ADAMS Accession No. ML15091A169).

Comments Associated with March 13, 2015, Public Meeting

Enclosures 2 and 3 of this document provide a listing of the verbal and written comments received, respectively, as a result of the March 13, 2015, public meeting. The meeting summary was issued on April 13, 2015 (ADAMS Accession No. ML15077A410).

Enclosure 2 documents NRC staff's response to the 41 verbal comments provided during the public meeting. The comments were generally focused on the spent fuel storage and transportation and decommissioning portions of the NRC staff's draft white paper. Many of the comments involved the independent spent fuel storage installations found at those power reactors that recently permanently ceased operation and the decommissioning of the same reactors. As a result of these comments, NRC staff added discussions to the spent fuel storage and decommissioning portions of the Enclosure 1 assessment to provide additional basis for its conclusions in these areas.

NRC staff also addressed 14 sets of email comments provided as a result of the public meeting. Enclosure 3 provides a summary of these comments and the NRC staff's response to each. As documented in Enclosure 3, NRC staff considered comments on the NRC staff's draft white paper from the licensees for the Massachusetts Institute of Technology research reactor (MITR), the University of Missouri at Columbia research reactor (MURR), and the National Bureau of Standards and Technology test reactor (NBSR). The comments from these licensees mainly involved clarifications of the various RTR attributes and capabilities to respond to accidents. Other email comments included concerns about the NRC staff's assessment of spent fuel storage at power reactors that have ceased operations and the NRC staff's assessment of other decommissioned facilities. NRC staff made changes to the Enclosure 1 assessment to address these comments, as appropriate.

Summary of Results:

For the eight classes of licensees discussed in the enclosure, the NRC staff's overall conclusion is that existing regulatory processes provide for adequate protection of public health and safety.

NRC staff's evaluation found two areas where work will continue to ensure that Fukushima lessons learned are appropriately considered:

- NRC staff will continue its efforts to resolve concerns with fuel facilities' safety assessments and the supporting documentation with respect to the treatment of natural phenomena hazards. NRC staff expects to issue a generic letter on this subject in 2015.

- For the three non-power reactors that have thermal power ratings in excess of 2 Megawatt thermal (MW_t), NRC staff is performing further assessments to determine if regulatory actions are necessary to mitigate certain beyond-design-basis external events.

Fuel Cycle Facility Assessment

As discussed in SECY-11-0137 and SECY-13-0095, "Fourth 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami" (ADAMS Accession No, ML13213A304), NRC staff completed inspections at selected fuel facilities and the results were used to perform a systematic evaluation of the processes and regulations applicable to fuel facilities. As a result of this evaluation, NRC staff concludes that the current regulatory approach and requirements of these licenses continue to serve as a basis for reasonable assurance of adequate protection of public health and safety. However, as described in greater detail in Enclosure 1 of this document, for the Honeywell Metropolis Works Facility, the inspections found potentially significant safety issues and NRC staff took immediate steps to ensure corrective actions were taken. In addition, NRC staff found generic issues regarding compliance with the current regulatory framework with regard to the treatment of certain natural phenomena events in the facilities' (uranium conversion, enrichment, and fuel fabrication) safety assessments. As discussed in SECY-15-0045, "Issuance of Generic Letter 2015-01, 'Treatment of Natural Phenomena Hazards in Fuel Cycle Facilities,'" dated March 27, 2015 (ADAMS Accession No. ML15036A141), NRC staff is developing a generic letter to request information from licensees that was not readily available during the inspections, to verify that compliance is being maintained with regulatory requirements and license conditions regarding the treatment of natural phenomena hazards.

Research and Test Reactor Assessment

NRC staff's review associated with the three highest-powered RTRs (MITR, MURR, and NBSR) identified the need for additional limited-scope assessments. For these three reactors, NRC staff will perform additional assessments regarding the reactors' capabilities to prevent or mitigate loss of coolant accidents as a result of beyond-design-basis natural phenomena (e.g., seismic events). This is because for these RTRs, the early loss of reactor coolant can result in failure of the fuel cladding and subsequent radiological release unless reactor coolant makeup can be provided from installed facility equipment or from portable external sources.

Both the MITR and MURR reactors are tank type reactors. Because of their low power, natural convection flow of reactor coolant is sufficient to remove decay heat from these reactors and prevent bulk boiling, even in the event of a loss of all electrical power and active decay heat removal systems. Therefore, there is not a near-term need to replenish the water around the reactor fuel lost by evaporation. However, if the initiating external event also causes (or occurs concurrently with) the failure of the core tank and the loss of the reactor pool integrity, then the resulting loss of coolant inventory could result in inadequate decay heat removal and fuel damage.

NBSR is the highest-powered RTR licensed by the NRC at 20 MW_t. In the case of NBSR, if there is an extended loss of electrical power to operate the active decay heat removal systems or damage to the active decay heat removal system that prevents its use, fuel damage could occur unless the decay heat removal systems are restored or reactor coolant makeup using portable external sources can be initiated to make up for evaporation and boil-off. As such, in addition to performing additional assessments regarding NBSR's capabilities to prevent or mitigate loss of coolant accidents as a result of beyond-design-basis natural phenomenon (e.g., seismic events), NRC staff will also assess the NBSR's capabilities to prevent or mitigate loss of decay heat removal capabilities due to an extended loss of alternating current power (e.g., that could result from an extreme flood or other extreme natural phenomenon).

The additional assessments for MURR, MITR, and NBSR will be performed by NRC staff with expertise in the areas of seismology, hydrology, structural analysis, and RTR design and operation. These assessments are expected to be limited in nature, will use existing budgeted resources, and will be prioritized considering other high-priority work that is being performed by these staff. Upon completion, NRC staff will provide a Commissioners' Assistants Note presenting the results of the assessments and, if appropriate, a discussion of the work necessary to resolve any remaining concerns at these facilities.

CONCLUSION:

For the majority of regulated facilities discussed in this paper, NRC staff has determined that no other analysis or regulatory action is needed based on its review of the lessons learned from the Fukushima Dai-ichi accident. If responses to the fuel facility generic letter or the further assessments of the three highest-powered RTRs cause NRC staff to determine that additional regulatory actions are needed to address Fukushima lessons learned, then staff will interact with stakeholders and, as appropriate, engage the Commission.

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Enclosures:
1. Applicability of Fukushima Lessons Learned to Facilities Other Than Operating Power Reactors
2. Discussion of Public Comments Received During March 13, 2015, Meeting
3. Discussion of Email Comments Associated with Staff's Draft Assessment

APPLICABILITY OF FUKUSHIMA LESSONS LEARNED TO FACILITIES OTHER THAN OPERATING POWER REACTORS

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Acronyms

ac	alternating current
ACI	American Concrete Institute
ADAMS	Agencywide Documents Access and Management System
ALARA	as low as is reasonably achievable
AMP	aging management program
AV	Apparent Violation
BWR	boiling water reactor
CFR	*Code of Federal Regulations*
DBA	design basis accident
DOE	Department of Energy
DOT	Department of Transportation
DP	decommissioning plan
EP	emergency plan
EPA	Environmental Protection Agency
EPZ	Emergency Planning Zone
FBI	Federal Bureau of Investigation
FEMA	Federal Emergency Management Agency
FEP	facility effective power
GTCM	greater than critical mass
IAEA	International Atomic Energy Agency
IMC	Inspection Manual Chapter
IN	Information Notice
ISFSI	Independent Spent Fuel Storage Installation
ISR	in situ recovery
LEU	low-enriched uranium
LIP	local intense precipitation
LLW	low-level waste
LOCA	loss of coolant accident
LTP	license termination plan
LRWPA	Low-level Radioactive Waste Policy Amendments Act of 1985
MHA	maximum hypothetical accident
MIT	Massachussetts Institute of Technology
MOX	mixed oxide
MURR	University of Missouri at Columbia research reactor
MW_t	megawatts-thermal
NBSR	National Institute of Standards and Technology test reactor
NEA	Nuclear Energy Agency
NPH	natural phenomena hazard
NPR	nonpower reactors
NTTF	Near Term Task Force
NRC	Nuclear Regulatory Commission
NSTS	National Source Tracking System
NYSERDA	New York State Energy Research and Development Authority
PMF	probable maximum flood
PSDAR	post-shutdown decommissioning activities report
RASSC	Radiation Safety Standard Committee

Acronyms (continued)

RWMC	Radiation Waste Management Committee
RTR	research and test reactors
SAR	Safety Analysis Report
SBO	station blackout
SSCs	structures, systems, and components
TI	temporary instruction
TLAA	time-limited aging analysis
UF_6	uranium hexaflouride
UFSAR	Updated Final Safety Analysis Report
USEC	United States Enrichment Corporation
USGS	United States Geological Survey
URI	Unresolved Item
WASSC	Waste Safety Standards Committee
WVDP	West Valley Demonstration Project
WVDPA	West Valley Demonstration Project Act of 1980

1. Spent Fuel Storage and Transportation Systems

I. Current Regulatory Framework

Radiation exposure is the main hazard associated with spent fuel storage facilities, systems, and transportation packages. These facilities, systems, and packages have multiple components and use specific materials of construction to protect workers, the public, and the environment. The U.S. Nuclear Regulatory Commission's (NRC's) responsibility is to review, approve, and inspect the design of storage facilities, systems, and transportation packages to ensure the safe and secure storage and transport of spent nuclear fuel. The U.S. Department of Transportation (DOT) is the United States' competent authority for defining requirements for radioactive materials in transit.

The following sections summarize the existing regulatory framework to ensure that radioactive materials under NRC's authority are stored and transported safely and securely.

<u>Licensing</u>

The regulatory infrastructure for licensing spent fuel storage (Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72)[1] and transportation package designs (10 CFR Part 71)[2] allows for issuing certificates of compliance, general licenses, or specific licenses. Certificates of compliance are granted to vendors for a particular storage cask or transportation system design. The following sections describe the NRC's licensing requirements for the safe and secure storage and transportation of spent nuclear fuel.

a. <u>Storage of Spent Nuclear Fuel</u>

The NRC's responsibility is to review and approve the design of storage systems and facilities for spent nuclear fuel and radioactive material in accordance with applicable regulations and guidance. Dry storage systems perform the safety functions of ensuring subcriticality, and confining and shielding the radioactive contents during the period of storage. For the design of the dry cask storage systems, vendors are required to consider natural phenomena hazards (NPHs) based on the 10 CFR Part 72 regulations. Vendors perform bounding analyses to verify the adequacy of the dry cask storage systems to comply with the pertinent regulations. Upon review of the applicant's safety analysis report, if appropriate safety and security margins are maintained, the results of the review are documented in a safety evaluation report. Then, the NRC issues a certificate of compliance accompanied with technical specifications to the cask vendor.

A utility that has an existing 10 CFR Part 50 or Part 52 license can have a general or specific license to store spent nuclear fuel (i.e., an independent spent fuel storage installation). For a general license, the utility is responsible for verifying that the design parameters of the selected cask system bound the NPHs that may occur at their site. A

[1] 10 CFR Part 72, "Licensing requirements for the independent storage of spent nuclear fuel and high-level radioactive waste, and reactor- related greater than Class C waste."
[2] 10 CFR Part 71, "Packaging and transportation of radioactive material."

specific license is required for any entity that does not already hold a 10 CFR Part 50 or 10 CFR Part 52 license or for any utility with site-specific parameter(s) that are not bounded by the certified vendor's designs. The applicants that apply for a specific license must verify the adequacy of the cask design to withstand the specified NPHs for the given site.

The United States has 55 general licensees and 15 specific licensees of independent spent fuel storage installations. The two types of independent spent fuel storage installations are dry storage or wet storage (i.e., spent nuclear fuel pool). There is only one wet storage facility licensed under 10 CFR Part 72 in the United States, G.E. Morris, and its fuel is more than 20 years old.[3]

Initial cask certifications and specific licenses were issued for a 20-year term. In 2011, NRC revised the requirements in 10 CFR Part 72 to allow for initial license and certification periods of up to 40 years. The regulations also allow for 40-year renewals of both the specific license and cask certifications. The requirement for renewal of specific licenses is contained in 10 CFR 72.42, "Duration of license; renewal." This regulation specifies that the renewal application include:

(1) Time-limited aging analyses (TLAAs) that demonstrate that structures, systems, and components important to safety will continue to perform their intended function for the requested period of extended operation; and

(2) A description of the aging management program (AMP) for management of issues associated with aging that could adversely affect structures, systems, and components important to safety.

The requirements for storage cask certification renewals are contained in 10 CFR 72.240, "Conditions for spent fuel storage cask renewal." Analyses of aging effects and an associated aging management program are also required for cask certification renewal.

NRC staff reviews renewal applications to ensure the independent spent fuel storage installation will continue to provide safe and secure storage of spent nuclear fuel throughout the renewal period. The performance of the storage system in both the initial and renewal terms must meet all regulatory requirements. This evaluation considered storage casks whether in their initial term or a renewed license or certification term.

b. Transportation of Spent Nuclear Fuel

The NRC's regulatory authority for the transport of radioactive material is delineated in the memorandum of understanding with DOT[4] and 10 CFR 71.5, "Transportation of licensed material." DOT is the competent authority for all transportation matters in the

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[3] NRC Inspection Report 072-00001/11-01, dated March 13, 2011.

[4] Memorandum of Understanding between the U.S. Department of Transportation and the U.S. Nuclear Regulatory Commission published on July 2, 1979; 44 FR 38690.

United States. The NRC's responsibility is to ensure that packages for fissile materials and Type B quantities[5] are designed, manufactured, and handled safely, as well as transported securely. Packages approved by the NRC shall comply with 10 CFR Part 71 and can be used at any place in the United States.

Domestically, NRC supports DOT on the review of packages that would be used in the United States that have been approved in other countries for transporting radioactive material. This review process is called revalidation. During the revalidation process, the NRC determines if the package complies with the International Atomic Energy Agency (IAEA) regulatory requirements for transportation of radioactive materials.[6]

Internationally, NRC participates in forums related to the transportation and storage of radioactive materials. The NRC also assesses if changes should be made to 10 CFR Part 71, when IAEA regulations are revised.

Inspection and Oversight

NRC staff inspects certificate holders and independent spent fuel storage installations. NRC staff inspects certificate holders to ensure compliance with quality assurance requirements for manufacturing and handling storage system or package designs. The staff also inspects the operational aspects of the independent spent fuel storage installations and spent fuel cask loading campaigns.

Emergency Preparedness

a. Storage of Spent Nuclear Fuel

Contents of an emergency plan (EP) for spent fuel storage facilities are delineated in 10 CFR 72.32. Per 10 CFR 72.32, an application for a specific-license for an independent spent fuel storage installation must include an EP if the facility is:

1. not located:

   i. on the site of a nuclear power reactor, or

   ii. within the exclusion area as defined in 10 CFR Part 100, "Reactor site criteria," of a nuclear power reactor, or

2. located on the site of a nuclear power reactor:

   i. which does not have an operating license, or

   ii. that is not authorized to operate.

[5] A Type B quantity is defined in 10 CFR 71.4, "Definitions."
[6] TS-R-1, "Regulations for the Safe Transport of Radioactive Material, 2009 Edition Safety Requirements."

For independent spent fuel storage installations located on the site or within the exclusion area of a nuclear power reactor licensed for operation, 10 CFR 72.32(c) specifies that "the EP required by 10 CFR 50.47 shall be deemed to satisfy" the 10 CFR 72.32 requirements for an EP. There are no current regulatory requirements for establishing emergency planning zones in the EP for independent spent fuel storage installations.

b. Transportation of Spent Nuclear Fuel

As previously mentioned, the NRC and the DOT share responsibility for regulating the transportation of radioactive materials. The NRC is responsible for ensuring that escorts are properly trained to respond to events as specified in 10 CFR Part 73, Appendix D,[7] including severe weather conditions and security events. The DOT defines requirements for the shippers of radioactive materials. Regulatory requirements for shippers related to emergency response are specified in 49 CFR Part 172.

Security

Existing NRC regulations include physical security requirements for facilities with a general or specific license for an independent spent fuel storage installation. The NRC and the DOT share regulatory responsibility for the safe transport of shipments.

a. Storage of Spent Nuclear Fuel

Licensees with either a specific or general license for an independent spent fuel storage installation must maintain their security systems in an operable condition. This includes measures to maintain security systems available during power losses per 10 CFR 73.51 and 10 CFR 73.55. Generally-licensed independent spent fuel storage installation security requirements are found in 10 CFR 72.212 and 10 CFR 73.55. The security requirements for a specific license of independent spent fuel storage installations are found in 10 CFR 72.180–194 and in 10 CFR 73.51.

After the terrorist attacks of September 11, 2001, the NRC issued security orders to all spent nuclear fuel storage facilities to impose greater security measures beyond the requirements in the NRC regulations at the time. The NRC continues to issue security orders to independent spent fuel storage installations and is revising 10 CFR Part 73 to codify the requirements of the security orders.

b. Transportation of Spent Nuclear Fuel

NRC's security requirements for spent nuclear fuel in transit include:

1. training requirements for escorts

[7] 10 CFR Part 73, "Physical protection of plants and materials," Appendix D, "Physical protection of irradiated reactor fuel in transit, training program subject schedule."

2. review and approval of shipment routes of spent nuclear fuel [8]
3. requirements for personnel accessing the shipment[9]

II. Post-Fukushima Event Evaluations/Assessment

NRC staff evaluated both transportation systems and storage designs to determine the effect of NPHs and the applicability of the Near Term Task Force (NTTF) recommendations. For the transportation assessment, the staff used NUREG-2125, "Spent Fuel Transportation Risk Assessment,"[10] published in January 2014, as the basis for the evaluation.

Also, NRC staff participated in a series of table top exercises to evaluate the effect of selected NPHs to existing storage designs. NRC staff did not evaluate site-specific designs, because the main design characteristics[11] of storage systems were the same for general and specific licensees. NRC staff's expertise included knowledge of 10 CFR Part 72 requirements to evaluate the adequacy of the storage design in the following areas:

- confinement evaluation
- criticality safety evaluation
- materials evaluation
- shielding evaluation
- structural evaluation
- thermal evaluation

The staff divided its qualitative assessment into the following phases:

Phase 1.	Determine initial applicability of NTTF recommendations to spent fuel storage and transportation certificate holders and licensees.
Phase 2.	Determine commonalities among the design characteristics of the storage systems. (Note: This phase did not apply to the NRC staff's assessment of transportation of spent fuel).
Phase 3.	Determine qualitative magnitude of consequences of NPHs and external events. (Rely on NUREG-2125 as basis for the transportation qualitative assessment.)
Phase 4.	Document findings, determine path forward, and develop input.

NRC staff's qualitative assessment is based on a single canister and assumes that all spent fuel systems and operations are in compliance with 10 CFR Part 72 regulations. Loading activities in the pool are performed in accordance with the licensee's 10 CFR Part 50 (or 10 CFR Part 52) safety programs. Final closure and on-site transfer are performed in accordance with technical specifications and site procedures. For welded canister-based systems, the transfer cask

[8] 10 CFR 73.37, "Requirements for physical protection of irradiated reactor fuel in transit"
[9] 10 CFR Part 73.38, "Personnel access authorization requirements for irradiated reactor fuel in transit"
[10] ADAMS Accession No. ML14031A323.
[11] I.e., wet storage or dry storage, horizontal or vertical system orientation, and welded or bolted closure.

provides additional protection to the storage canister during on-site transfer, while for directly loaded systems, the cask is fully prepared for storage before on-site transfer.

a. Phase 1. Determine initial applicability of NTTF recommendations to spent fuel storage and transportation certificate holders and/or licensees.

The NTTF report, "Recommendations for Enhancing Reactor Safety in the 21st Century," has 12 recommendations. NRC staff considered these recommendations for applicability to independent spent fuel storage installations and the transport of spent nuclear fuel and evaluated the recommendations to determine if any future action is warranted. The following table summarizes the result of the review.

Near-Term Task Force Recommendations and Future Actions

Recommendations		**Review Result**
1	Establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.	No Action Needed.[12] No regulatory gaps were identified as a result of the staff's initial review. The decision was confirmed after additional assessment (discussed below) was completed.
2	The task force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection of structures, systems and components (SSCs).	No Action Needed. Based on initial review and confirmed by bounding analysis discussed below, the staff did not identify a need for additional analysis in this area.
3	The task force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	No Action Needed. As discussed below, ISFSIs are made of material that is not flammable, and bounding flooding analysis shows no need for additional analysis. Moreover, combustible loads are currently minimized at ISFSIs. For the one wet storage facility in the U.S., the stored fuel is more than 20 years old. This wet storage facility is a below-ground reinforced concrete pool with a welded stainless steel liner. Based on a review of multiple external scenarios (as discussed below), NRC staff determined that fuel melt is not credible as a result of these scenarios.
4	The task force recommends that the NRC strengthen station blackout (SBO) mitigation capability at all operating and new reactors for design-basis and beyond-design-	No Action Needed. ISFSIs remove decay heat by passive mechanisms. The use of power is limited to monitoring systems. Loss of power will not lead to fuel damage. The fuel at the existing wet storage facility is more than 20

[12] No Action. No further study or regulatory action by NRC staff is warranted.

	Recommendations	Review Result
	basis external events.	years old and does not have a near term need for active decay heat removal systems.
5	The task force recommends requiring reliable hardened vent designs in boiling-water reactor (BWR) facilities with Mark I and Mark II containments.	Not Applicable.[13] There are no credible scenarios that create hydrogen in quantities of concern for spent fuel storage and transportation systems.
6	The task force recommends, as part of the longer term review, that the NRC identify insights about hydrogen control and mitigation inside containment or in other buildings as more information is revealed through further study of the Fukushima Dai-ichi accident.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for spent fuel storage and transportation systems.
7	The task force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	No Action Needed. The fuel at the U.S. wet storage facility is more than 20 years old and does not have a near term need for active decay heat removal systems or near term makeup.
8	The task force recommends strengthening and integrating onsite emergency response capabilities such as EOPs [emergency operating procedures], SAMGs [severe accident management guidelines], and EDMGs [extensive damage mitigation guidelines].	No Action Needed. Fukushima lessons learned for power reactors showed prolonged loss of AC power and multiunit events could have severe consequences leading to a need to update EP response. Prolonged loss of AC power or multiunit event is not a concern for passively cooled ISFSIs or the existing wet storage facility with fuel more than 20 years old. No need to upgrade existing EP requirements for spent storage and transportation systems was identified.
9	The task force recommends that the NRC require that facility EPs address prolonged SBO and multiunit events.	No Action Needed. See item 8 above.
10	The task force recommends, as part of the longer term review, that the NRC should pursue additional EP topics related to multiunit events and prolonged SBO.	No Action Needed. See item 8 above.
11	The task force recommends, as part of the longer term review, that the NRC should pursue EP topics related	No Action Needed. See item 8 above.

[13] Not applicable. The NTTF recommendation for commercial nuclear power reactors does not apply to this "other regulated facility."

	Recommendations	Review Result
	to decision-making, radiation monitoring, and public education.	
12	The task force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the reactor oversight process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.	No Action Needed. No regulatory gaps were identified as a result of the staff's initial review. The decision was confirmed after additional assessment (discussed below) was completed.

b. <u>Phase 2. Determine commonalities among the design characteristics of the storage systems (Note: this phase did not apply to NRC staff's assessment of transportation of spent fuel)</u>

Based on the current regulatory framework for licensing, independent spent fuel storage installations were categorized into two major groups, general licenses and specific licenses.[14] NRC staff further grouped the casks systems into different design characteristics to simplify the qualitative assessment. Design characteristics included:

1. orientation on the independent spent fuel storage installation pads (i.e., vertical, horizontal, or underground)
2. closure type (i.e., bolted or welded)[15]
3. vents on overpack (i.e., vented or not vented)

After grouping the storage systems, NRC staff qualitatively evaluated the effects of selected natural phenomena and external events.

c. <u>Phase 3. Determine qualitative magnitude of consequences of NPHs and external events.</u>

NRC staff qualitatively evaluated the radiological consequences of the natural phenomena and external events. The magnitudes of the consequences were defined as follows:

1. Low – No radiation-related deaths or injuries expected; no offsite contamination.

2. Medium – Few radiation-related deaths or injuries; little to no offsite contamination.

[14] Each type of licensee uses a dry cask storage system to place the fuel removed from the spent fuel pool. Then, these casks systems are placed on the independent spent fuel storage installation concrete pad.
[15] Vertical systems only.

3. High – Significant radiation-related deaths or injuries and significant offsite contamination or property damage.

NRC staff's initial screening found that natural phenomena such as lightning, snow and ice loads, external fire, extreme temperature, and drought were either determined to be not applicable or low to no consequence events. Dry cask storage systems are passive and designed for the heat loads associated with spent nuclear fuel. Natural phenomena of extreme cold or heat (low or high ambient temperatures) will not impact the safety performance of the storage system such that confinement of the system would fail. Similarly, lightning and fire would not negatively impact the safety performance of the storage system. The storage system is comprised of materials that are not flammable (concrete and steel). The presence of flammable materials, like diesel fuel, is limited by technical specification around the dry storage system, however, the temperatures associated with fire would not impact the containment function of the storage system. Drought is not applicable because there is no use of water in dry storage. NRC staff performed more in-depth assessment for the following natural phenomena and external events for dry storage cask systems:

1. seismic
2. flooding
3. high winds (e.g., hurricanes),
4. tornado winds and tornado missiles, and
5. loss of off-site power

The regulations in 10 CFR Part 72 require the applicants to assess natural phenomena events as part of the safety basis for a storage facility or container design. For this assessment, NRC staff considered the potential impacts of natural phenomena that would be beyond the design basis evaluation that would be required to meet 10 CFR Part 72. Based on the qualitative evaluation, NRC staff determined that these events also would result in low radiological consequences. The following summarizes the NRC staff's qualitative evaluation:

1. **Seismic** – Per 10 CFR Part 72.102[16] and 10 CFR Part 72.103[17] the design-basis earthquake for dry cask storage systems varies depending on the user's location of the storage cask design. Nevertheless, for a storage system, the consequences of an earthquake are bounded by a hypothetical event called "non-mechanistic tip-over." Dry casks storage systems are designed to demonstrate structural adequacy under this hypothetical scenario. This event is independent of any natural phenomena events, and the structural integrity of the

[16] 72.102, "Geological and seismological characteristics for applications before October 16, 2003, and applications for other than dry cask modes of storage."
[17] 72.103, "Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003."

fuel and the cask body is verified for their adequacy to withstand the potential of a cask tipping over and hitting the concrete supporting pad.[18]

2. **Flood** – NRC staff considered scenarios such as a (1) storage system fully submerged in water, (2) partial flooding around the storage system, and (3) blockage of the overpack vents for those canister-based systems with natural convection cooling vented overpacks. The worst case scenario for loss of heat transfer is a partial flood around the storage system. This scenario would not cause a radioactive release to the environment because the temperature would not exceed allowable temperatures for the materials of construction for system confinement.

   For the third scenario, NRC staff assumed that all vents of an overpack system were blocked with debris, the debris was not removed, and the temperature of the fuel increased (i.e., cooling capability of the system decreased). The results of the steady-state thermal analysis for a vertical cask system and a horizontal cask system indicate that a release from the confinement is not expected to occur. The heat load in the cask is at its highest when initially placed into storage and the system cools overtime. NRC staff considered the highest allowable heat loads in the cask system for this evaluation.

3. **High Winds** – For this scenario, NRC staff considered hurricane force winds. Regulatory Guide 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," and NUREG/CR-7005, "Technical Basis for Regulatory Guidance on Design-Basis Hurricane Wind Speeds for Nuclear Power Plants," are NRC's guidance used for addressing hurricane winds in the design of a nuclear power plant. Most of the independent spent fuel storage installations are collocated with a nuclear power plant; therefore, the same guidance applies. The hurricane wind speeds in NUREG/CR-7005 were based on a model developed for the American Society of Civil Engineers.[19]

   Based on NUREG/CR-7005, the resulting wind speeds are nominal 3-second peak-gust values at a height of 33 feet in flat open terrain. Using Regulatory Guide 1.221, the wind speed may vary from 130 mph (Texas) to 290 mph (Southeastern Atlantic Coastline) depending on the independent spent fuel storage installation's location. Missiles generated by hurricanes were not considered in this portion of the assessment, because the tornado assessment would bound high winds generated-missile events. Given the size and weight of spent fuel storage systems, it is not anticipated that high winds will displace or overturn the cask; however, the worst case in a beyond design basis event due to high winds would be bounded by the "non-mechanistic tip-over" which is described above for the seismic evaluation.

[18] Independent spent fuel storage installation pad.
[19] ASCE/SEI 7-05, "Minimum Design Loads for Buildings and Other Structures."

4. **Tornado Winds and Tornado Missiles** – Dry cask storage systems' structures, systems, and components that are important to safety must be designed to withstand the effects of natural phenomena hazards without losing the capability to perform their safety function. NRC Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, March 2007, offers guidance that NRC staff considers acceptable in selecting the design-basis tornado and tornado-generated missiles for a nuclear power plant. Vendors of dry cask storage systems use these criteria to comply with the 10 CFR Part 72 requirements.

   During the qualitative assessment, NRC staff considered the potential for damage to the cask confinement boundary for the bolted system due to a beyond design basis tornado missile event. Given the robustness of the spent fuel cask design, this would not result in a significant release or increase in the radiation levels in the vicinity of the cask. The consequence of this event is categorized as low since there would be no off-site contamination.

5. **Loss of Off-Site Power** – The surveillance systems at the independent spent fuel storage installations that would need power to operate are pressure sensors and security systems. During the qualitative assessment, NRC staff found that the loss of power of pressure sensors would not cause a release of radioactive material. Requirements of security surveillance systems are included in 10 CFR Part 73 as discussed above.

There is only one wet storage independent spent fuel storage installation in the United States (i.e., G.E. Morris) and the stored fuel is more than 20 years old.[20] The G.E. Morris facility is a below-ground reinforced-concrete pool with a welded stainless steel liner. This facility was inspected in March 2011,[21] after the event at Fukushima. NRC inspectors reviewed scenarios, such as station blackout, seismic, tornado, flood, and fire. NRC staff found that the fuel would not melt as a result of these events because of its limited heat load.

NRC staff considered the findings and conclusions documented in NUREG-2125, "Spent Fuel Transportation Risk Assessment" (ADAMS Accession No. ML12125A218), in evaluating the consequence of the NPHs with respect to the transport of spent nuclear fuel. NUREG-2125, published in January 2014, estimated the risk associated with the transportation of spent nuclear fuel by examining the behavior of three NRC-certified casks during routine transportation and in transportation accidents. NRC staff's qualitative evaluation focused on comparison of the impact and fire accident scenarios in NUREG-2125 to the NPH. Flooding is not an accident scenario considered in NUREG-2125 because of the regulatory requirement for spent fuel transportation packages to be designed and constructed and the contents limited such that it would be subcritical if water were to leak into the containment system such that maximum reactivity of the fissile

[20] NRC Inspection Report 072-00001/11-01, dated March 13, 2011.
[21] NRC Inspection Report 072-00001/11-01, dated March 13, 2011.

material would be attained (10 CFR 71.55, "General requirements for fissile material packages"). Therefore, flood is already an analyzed condition for spent fuel transportation packaging. The staff concluded that the NPHs are bounded by the accident scenarios evaluated in NUREG-2125 and would have low consequence.

III. International Experience

Since the March 11, 2011, seismic and tsunami events at the Fukushima Dai-ichi nuclear power plants in Japan, NRC has been actively involved in information exchanges with the international community at conferences and forums such as the Structural Mechanics in Reactor Technology and IAEA. NRC staff has coordinated and chaired technical sessions to address issues related to the safe storage and transportation of spent nuclear fuel as well as presented papers at these forums. The insights gained during these technical exchanges have been used, as applicable, for facilities within the United States. NRC staff has followed the progress by IAEA on the Action Plan on Nuclear Safety (GOV/2011/59-GC(55)/14) and the results of the IAEA gap analysis, as it relates to spent fuel storage and transportation. NRC staff will continue to participate on working groups such as the IAEA safety guide SSG-15: Storage of Spent Nuclear Fuel and SSR-6: Regulations for the Safe Transport of Radioactive Material, are evaluated for revision. Any enhancements that are identified in these IAEA documents will be considered for applicability to the NRC's regulations for spent fuel storage and transportation.

IV. Conclusions and Recommendations

NRC staff found that the NRC's existing regulatory framework ensures the safe and secure storage and transportation designs for radioactive material licensed by the NRC. Based on the qualitative assessment, NRC staff did not find safety concerns associated with the designs of spent fuel storage and transportation systems.

The defense-in-depth philosophy is embedded in 10 CFR Part 71 and 10 CFR Part 72 regulations. NRC staff ensures that designs proposed by applicants, licensees, and certificate holders consider industry standards and include layers of protection to maintain doses within the regulatory limits. The systems are built and handled under an NRC approved quality assurance program. In term of oversight of NRC approved systems, NRC staff has guidance to inspect and schedule inspections to ensure that storage facilities, storage systems, and transportation systems are built, and operate, as specified in the design approved by the NRC.[22] Therefore, NRC staff concludes that no further regulatory action or study is necessary.

[22]Inspection Manual Chapter 2690, "Inspection Program for Dry Storage of Spent Reactor Fuel at Independent Spent Fuel Storage Installations and for 10 CFR Part 71 Transportation Packagings."

2. Fuel Facilities

I. Overview of Fuel Cycle Facilities, Academic and Other Institutions

Fuel cycle facilities involved in conversion, enrichment, and fuel fabrication are regulated through a combination of regulatory requirements; licensing; safety oversight (including inspection, assessment of performance, and enforcement); evaluation of operational experience; and regulatory support activities. These facilities turn the uranium that has been removed from ore (as yellowcake) into fuel for nuclear reactors. In this process, the conversion facility converts yellowcake into uranium hexafluoride (UF_6). Next, an enrichment facility heats the solid UF_6 enough to turn it into a gas, which is "enriched," or processed to increase the concentration of the uranium-235 (U_{235}). Then the enriched uranium is manufactured into pellets. These pellets are placed into fuel assemblies and ultimately into nuclear reactors. <u>Fuel Cycle Process, Facilities, and Associated Hazards</u>

The regulations governing the licensing and operation of fuel cycle facilities are found below. Each type of facility presents different levels of risk to both the worker and offsite from events involving chemical and radiological material.

- 10 CFR Part 20, "Standards for protection against radiation"
- 10 CFR Part 21, "Reporting of defects and noncompliance"
- 10 CFR Part 40, "Domestic licensing of source material"
- 10 CFR Part 70, "Domestic licensing of special nuclear material"
- 10 CFR Part 73, "Physical protection of plants and materials"
- 10 CFR Part 74, "Material control and accounting of special nuclear material"
- 10 CFR Part 76, "Certification of gaseous diffusion plants"

A. <u>Uranium Conversion</u>

<u>Process</u>: After the yellowcake is produced at the mill, the next step is conversion into pure UF_6 gas suitable for use in enrichment operations. During this conversion, impurities are removed and the uranium is combined with fluorine to create the UF_6 gas. The UF_6 is then pressurized and cooled to a liquid. In its liquid state it is drained into 14-ton cylinders where it solidifies after cooling for approximately five days. The UF_6 cylinder, in the solid form, is then shipped to an enrichment plant.

<u>Hazards</u>: As with mining and milling, the primary risks associated with conversion are potential chemical and radiological events. Strong acids and alkalis are used in the conversion process, which involves converting the yellowcake (uranium oxide) powder to very soluble forms, leading to possible inhalation of uranium. In addition, conversion produces extremely corrosive chemicals that could cause fire and explosion hazards. Fire and explosion hazards are also a concern in areas where liquid UF_6 is stored and processed. When liquid UF_6 is released to the atmosphere, it reacts with the moisture in the air to form a dense vapor cloud that contains hydrogen fluoride gas (chemical hazard), a nonradioactive, extremely toxic substance.

<u>Plants</u>: Honeywell International Inc. Metropolis, Illinois.

B. Uranium Enrichment

Process: Enriched uranium is required in commercial light-water reactors to produce a controlled nuclear reaction. Enriching uranium increases the proportion of uranium atoms that can be "split" by fission to release energy (usually in the form of heat) that can be used to produce electricity. Not all uranium atoms are the same. When uranium is mined, it consists of about 99.3 percent uranium-238 (U_{238}), 0.7 percent uranium-235 (U_{235}), and less than 0.01 percent uranium-234 (U_{234}). The fuel for nuclear reactors has to have a higher concentration of U_{235} than exists in natural uranium ore. Normally, the amount of the U_{235} isotope is enriched from 0.7 percent of the uranium mass to about 5 percent.

Hazards: The principal hazards at an enrichment plant are the chemical hazards in handling UF_6. When UF_6 contacts moisture in air, it reacts to form hydrogen fluoride and uranyl fluoride. The chemical hazards of compounds of uranium in soluble form such as UF_6 and uranyl fluoride are much greater than the radiological hazards of those same compounds. In addition, hydrogen fluoride can be very dangerous if inhaled; inhalation is the principal hazard at an enrichment plant. These hazards are controlled by plant design and administrative controls to confine soluble uranium compounds. The radiological hazards are relatively low and containers of natural, enriched, and depleted uranium can be handled without extra shielding. Another hazard for this type of facility is the potential for mishandling the enriched uranium, which could create a criticality accident (inadvertent nuclear chain reaction).

Plants:

- Gaseous Diffusion Uranium Enrichment Facility
  - United States Enrichment Corporation (USEC) Inc. in Paducah, Kentucky (No longer a NRC-licensed facility)

- Gas Centrifuge Uranium Enrichment Facilities
  - American Centrifuge Plant, LLC (USEC) in Piketon, OH (License issued, construction halted)
  - Louisiana Energy Services in Eunice, NM
  - AREVA Enrichment Services Eagle Rock, LLC , Idaho Falls, ID (License issued, construction not started)

- Laser Separation Enrichment Facility
  - GE-Hitachi in Wilmington, NC (License issued, construction not started)

C. Uranium Fuel Fabrication

Process: Fuel fabrication facilities convert enriched UF_6 into fuel for nuclear reactors. Fabrication also can involve mixed oxide (MOX) fuel, which is a combination of uranium and plutonium components. The NRC regulates several different types of nuclear fuel fabrication operations, such as light water reactor low-enriched uranium fuel and light water reactor mixed oxide fuel.

Fuel fabrication for light water power reactors typically begins with receipt of low-enriched uranium (LEU) hexafluoride from an enrichment plant. The UF_6, in solid form in containers, is heated to gaseous form, and the UF_6 gas is chemically processed to form LEU dioxide (UO_2) powder. This powder is then pressed into pellets, sintered into ceramic form, loaded into zircaloy tubes, and constructed into fuel assemblies. Depending on the type of light water reactor, a fuel assembly may contain up to 264 fuel rods and have dimensions of 5 to 9 inches square by about 12 feet long.

MOX fuel differs from LEU fuel in that the dioxide powder from which the fuel pellets are pressed is a combination of UO_2 and plutonium oxide (PuO_2). Congress directed NRC to regulate the Department of Energy's (DOE's) fabrication of MOX fuel, which uses repurposed plutonium from international nuclear disarmament agreements.

Hazards: Chemical, radiological, and criticality hazards at fuel fabrication facilities are similar to hazards at enrichment plants.

Plants:

- Uranium Fuel Fabrication
  - Global Nuclear Fuel-Americas, LLC in Wilmington, NC
  - Westinghouse Electric Company, LLC in Columbia, SC
  - Nuclear Fuel Services, Inc. in Erwin, TN
  - AREVA NP, Inc. in Richland, WA
  - Babcock & Wilcox Nuclear Operations in Lynchburg, VA

- Mixed Oxide Fuel Fabrication
  - Shaw AREVA MOX Services, LLC in Aiken, SC (Construction Authorization issued, construction ongoing)

D. Uranium Hexafluoride Deconversion

Process: As U_{235} is extracted, converted, and enriched in the uranium recovery, conversion, and enrichment processes for use in fabricating fuel for nuclear reactors, large quantities of depleted uranium hexafluoride (DUF_6), or “tailings,” are produced. These tailings are transferred into 14-ton cylinders which are stored in large yards near the enrichment facilities. A process called "deconversion" is then used to chemically extract the fluoride from the DUF6 stored in the cylinders. This deconversion process produces stable compounds, known as uranium oxides, which are generally suitable for disposal as low-level radioactive waste.

Hazards: Chemical exposure is the dominant hazard at deconversion facilities because uranium chemical compounds and other chemical compounds (such as hydrogen fluoride) are hazardous at low levels of exposure.

Plant: International Isotopes in Hobbs, NM (license issued, construction not started)

E. Academic and Other Institutions

Academic and other institutions use radioactive material in classroom demonstrations, laboratory experiments, and research, and to offer health physics support to other institutional radioactive materials users. These facilities are licensed in accordance with 10 CFR Parts 30, 40, or 70 depending on the type of materials possessed. These programs may vary in size from large, broad-scope programs involving chemical, physical, biological engineering, and biomedical research, to small programs using only gas chromatographs or self-shielded irradiators.

Hazards: These licensees have limited amounts of materials and have demonstrated by NRC-approved evaluation that no member of the public will exceed the thresholds of the regulations in 10 CFR Part 70. That is not to say that accidents cannot occur with these licensees, but because of the limited amount of materials possessed and the primarily sealed nature of the material, the effect to the public and the environment is limited.

II. Post-Fukushima Event Evaluations/Assessment

NRC staff evaluated and inspected selected fuel cycle facilities to confirm that licensees complied with regulatory requirements and license conditions; and to evaluate their readiness under NPH events and other licensing bases events related to NPHs. NRC staff's assessment considered the NTTF recommendations to determine whether other regulatory actions by the NRC are warranted. This assessment included consideration of new seismic hazards information from the U.S. Geological Survey (USGS) for the central and eastern United States which was the subject of an NRC generic communication to fuel facilities in NRC Information Notice 2010-19, "Updated Probabilistic Seismic Hazard Estimates in Central Eastern United States" (ADAMS Accession No. ML102160735).

As discussed in SECY-11-0137 and SECY-13-0095, NRC staff completed inspections at selected fuel facilities and the results were used to perform a systematic evaluation of the processes and regulations applicable to fuel facilities. Because of the evaluation, NRC staff concludes that the current regulatory approach and requirements of these licensees continues to serve as a basis for reasonable assurance of adequate protection of public health and safety. However, as described in greater detail below, for the Honeywell Metropolis Works Facility, the inspections found potentially significant safety issues and the NRC staff took immediate steps to ensure corrective actions were taken. In addition, NRC staff found generic issues regarding compliance with the current regulatory framework with regards to the treatment of certain natural phenomena events in the facilities' (uranium conversion, enrichment and fuel fabrication) safety assessments. As discussed in SECY-15-0045, "Issuance of Generic Letter 2015-01, 'Treatment of Natural Phenomena Hazards in Fuel Cycle Facilities,'" dated March 27, 2015 (ADAMS Accession No. ML15036A141), NRC staff is developing a generic letter to request information from licensees that was not readily available during the inspections, to verify that compliance is being maintained with regulatory requirements and license conditions regarding the treatment of NPHs.

A. Uranium Conversion, Enrichment, Fuel Fabrication, and Deconversion Licensees

On March 31, 2011, NRC staff issued Information Notice (IN) 2011-08, "Tohoku–Taiheiyou–Oki Earthquake Effects on Japanese Nuclear Power Plants—for Fuel Cycle Facilities," (ADAMS Accession No. ML110830824) to inform addressees of the potential challenges associated with preventing or mitigating the effects of natural phenomena events. IN 2011-08 recommended that addressees review the information for applicability to their facilities and consider actions, as appropriate, to ensure that features and preparations necessary to withstand or respond to severe external events from natural phenomena (e.g., earthquakes, tsunamis, floods, tornadoes, and hurricanes) are reasonable and consistent with regulatory requirements.

From December 2011 through May 2012, NRC staff conducted inspection activities in accordance with Temporary Instruction (TI) 2600/015, "Evaluation of Licensee Strategies for the Prevention and/or Mitigation of Emergencies at Fuel Facilities" (ADAMS Accession No. ML12286A284). The NRC completed the TI in three phases. In the initial phase, NRC staff reviewed licensing documents, including the safety assessments and EPs. The second phase consisted of NRC inspectors evaluating accident prevention measures and emergency actions through onsite evaluations that focused on credible natural phenomena and loss of utilities that support onsite systems (e.g. electricity and water). The third phase involved assessing whether the strategies and equipment were effective to prevent or mitigate emergencies during selected beyond licensing basis natural events and the extended loss of utilities that support onsite systems. In the review of licensing basis events, the NRC considered the following NPHs: seismic, flooding, and high winds (caused by hurricanes or tornadoes). The NRC also evaluated onsite fires that may result from seismic related equipment failures. Particular attention was given to earthquakes and flooding because of recent events and significant advancements in the state of knowledge of these hazards.

In addition, during the TI 2600/015 inspections, the staff also considered operating experience on the implementation of mitigation strategies and emergency procedures used by licensees to cope with natural phenomena events. The staff performed interviews of licensee personnel and walk-downs of the facility to assess how the facility coped with the occurrence of previous natural phenomena events (such as floods or hurricanes).

After the implementation of TI 2600/015, the NRC determined that the evaluated facilities had established programs, procedures, and equipment to respond to licensing basis events involving fire, flooding, and loss of utilities. However, based on information obtained from the inspection activities, NRC staff found that the assumptions used by licensees in developing the independent safety analysis and other safety assessments are not clearly described and documented. The NRC primarily attributed this to the lack of available facility design information and significant variations in the level of detail and rigor of implementation in the facility safety assessments with regards to the treatment of natural phenomena events. Therefore, the NRC inspectors were unable to verify that these facilities complied with their licensing basis and regulatory requirements. The NRC inspectors opened unresolved items (URIs) to further assess whether the evaluated licensees complied with license conditions, the requirements of 10 CFR 70.61 and 10 CFR 70.62(c), regarding NPH event accident sequences. A summary of the results of the inspections can be found in the table below. NRC staff has determined that for all the facilities inspected (except the Honeywell Metropolis Works Facility) in consideration of inherent seismic capacity in facility structures, existing safety programs in place, and radiological/chemical source terms, continued operation does not pose an imminent risk to public health and safety.

For the Honeywell Metropolis Works Facility, the NRC determined that the site Emergency Response Plan underestimated the amount of UF_6 and hydrogen fluoride that could potentially be released during credible seismic events or tornadoes. Specifically, the inspectors found that the process equipment in the licensee's Feed Materials Building lacked seismic restraints, supports, and bracing that would ensure process equipment integrity during certain credible seismic events or tornadoes. The NRC issued a confirmatory order that required the licensee to demonstrate its SSCs relied on for safety were adequate for seismic and tornado events. The facility structure and internal components were significantly retrofitted to improve the performance under seismic and tornado events. Additional information is available in NUREG-0090, Volume 36, "Report to Congress on Abnormal Occurrences: Fiscal Year 2013" (ADAMS Accession No. ML14150A073).

Summary of TI 2600/015 results

	Facility	Summary of Issues Identified
Part 76	Paducah (ADAMS Accession No. ML12131A437)	• Tornadoes were not considered a credible event because of the return period chosen for the evaluation basis event. The team determined that a tornado could be considered a credible event for the site if newer data is used to evaluate the probability of occurrence. However, the consequences of a tornado event are bounded by other safety basis events.
Part 70	AREVA (ADAMS Accession No. ML12122A094)	• Unresolved Item (URI) 70-1257/2012-006-01 was opened to further evaluate whether the licensee complied with the requirements of 10 CFR 70.62(c) and 70.61 performance requirements regarding natural phenomena events accident sequences.
	Babcock & Wilcox Nuclear Operations Group (ADAMS Accession No. ML12121A574)	• URI 70-27/2012-006-01 was opened to further evaluate whether the licensee complied with the requirements of 10 CFR 70.62(c) and 70.61 performance requirements regarding natural phenomena events accident sequences.
	Global Nuclear Fuel − Americas (ADAMS Accession No. ML12209A276)	• URI 70-113/2012-006-01 was opened to further evaluate whether the licensee complied with the requirements of 10 CFR 70.62(c) and 70.61 performance requirements regarding natural phenomena events accident sequences.
	Nuclear Fuel Services (ADAMS Accession No. ML12122A186)	• URI 70-143/2012-06-01 was opened to further evaluate whether the licensee complied with Table 2.2 of the license application regarding management measures for items relied on for safety PREP-A and PREP-B. • URI 70-143/2012-006-03 was opened to further evaluate whether the licensee complied with the requirements of 10 CFR 70.62(c) and 70.61 performance requirements regarding natural phenomena events accident sequences.

	Facility	Summary of Issues Identified
Part 40	Westinghouse – Columbia Fuels (ADAMS Accession No. ML12122A083)	• URI 70-1151/2011-07-01 was opened to review Westinghouse's response to the failure to ensure that the risk of an earthquake was limited by applying sufficient engineered controls, administrative controls, or both, to the extent needed so that, upon implementation of such controls, the event was highly unlikely. • URI 70-1151/2011-07-02 was opened to review Westinghouse's evaluation regarding whether all nuclear process under an earthquake were subcritical.
	Honeywell (ADAMS Accession No. ML12222A163)	• URI 40-3392/2012-006-01 was opened to evaluate whether the Metropolis Works Facility integrated safety analysis appropriately considered credible high consequence seismic and tornado events and subsequently designated plant features and procedures and management measures to ensure that the accident sequences (public and workers health and safety) remained highly unlikely or the consequences were mitigated to acceptable levels. • Apparent Violation (AV) 40-3392/2012-006-02 was identified for the failure to identify all relevant accident sequences related to credible seismic events and tornadoes that could result in large uranium hexafluoride (UF6) releases for which protective actions may be needed. • AV 40-3392/2012-006-003 was identified for the failure to supply complete and accurate information related to MTW's emergency response plan.

Note: Louisiana Energy Services was not inspected because it is a new facility designed and constructed with more demanding criteria as required by 10 CFR 70.64, "Baseline Design Criteria".

<u>Review of Near-Term Task Force recommendations</u>

NRC staff considered the 12 NTTF recommendations for applicability to fuel cycle facilities and evaluated to determine if any future action is warranted. The following table summarizes the result of the review.

Near-Term Task Force Recommendations and Future Actions

Recommendations		Review Result
1	Establish a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.

Recommendations		Review Result
2	The task force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection of SSCs.	No Immediate Action. Based on lessons learned from TI 2600/15 inspections staff identified a need to verify compliance regarding treatment of natural phenomena events (see Generic Letter discussion below).
3	The task force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	No Action Needed. Staff assessment noted inherent safety margin in fuel cycle facilities and NRC staff assessment considered new seismic hazards information from USGS which was provided to fuel facilities by NRC Information Notice 2010-19.
4	The task force recommends that the NRC strengthen SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events.	No Action Needed. Staff assessment considered the inherent safety margin in fuel cycle facilities, including but not limited to, existing safety programs in place and radiological and chemical source terms to determine that SBO mitigation capabilities do not need to be strengthened beyond those that already exist.
5	The task force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.
6	The task force recommends, as part of the longer term review, that the NRC points out insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-ichi accident.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.
7	The task force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	Not Applicable to fuel cycle facilities.
8	The task force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs.	No Action Needed. Fukushima lessons learned for power reactors showed prolonged loss of AC power and multiunit events could have severe consequences leading to a need to update EP response. No need to upgrade existing EP requirements based on inherent safety margin in fuel cycle facilities, including but not limited to, existing safety programs in place, and considering radiological and chemical source terms associated with fuel

	Recommendations	Review Result
		cycle facilities.
9	The task force recommends that the NRC require that facility EPs address prolonged SBO and multiunit events.	No Action Needed. See item 8 above.
10	The task force recommends, as part of the longer term review, that the NRC should pursue additional EP topics related to multiunit events and prolonged SBO.	No Action Needed. See item 8 above.
11	The task force recommends, as part of the longer term review, that the NRC should pursue EP topics related to decision-making, radiation monitoring, and public education.	No Action Needed. See item 8 above.
12	The task force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the reactor oversight process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.

Generic Letter: Treatment of Natural Phenomena Hazards in Fuel Cycle Facilities

Because of inspections, and as discussed in SECY-15-0045, NRC staff is proposing issuing a generic letter due to the generic applicability of the URIs across the nuclear fuel facility industry. Current NRC regulations require the evaluation of site hazards including natural phenomena events. The purpose of the generic letter is to request information from licensees to verify that compliance is being maintained with regulatory requirements and license conditions regarding the treatment of natural phenomena events.

NRC staff maintains engagement with industry and other stakeholders through public meetings and presentations during the annual Fuel Cycle Information Exchange. NRC staff plans to issue the generic letter in 2015 and expects to close out the review of the responses in 2016.

B. Academic and Other Institutions (Greater Than Critical Mass)

One of the considerations coming out of the fuel cycle review described above was to evaluate the readiness of greater than critical mass (GTCM) licensees as it applies to the lessons from the Fukushima accident. The readiness of GTCM facilities was evaluated by performing desktop reviews (ADAMS Accession No. ML14111A087). The recommendations of the NTTF were reviewed for applicability, as well as considerations found during the review of fuel cycle facilities. On the basis of this review, the continued operation of GTCM licensees do not pose an imminent risk to the public health and safety. The current regulatory approach and requirements of these licensees continues to serve as a basis for reasonable assurance of adequate protection of public health and safety. For these facilities, compliance with the current regulatory framework ensures that the consequences of any potential natural phenomena event are low to offsite receptors and workers.

III. International Experience

NRC staff has been actively involved in International forums with regards to discussions and lessons from the accident as it relates to fuel facilities. For example, NRC staff participated in an IAEA consultancy and technical meeting to develop a standard on the assessment of fuel cycle facilities in light of the Fukushima Dai-ichi accident. The standard aims to provide practical information and experiences on the performance of assessments of fuel cycle facilities.

In addition, NRC staff, in coordination with the Office of International Programs, met with the Director and Principal Inspector of the Decommissioning, Fuel, and Waste Program (Office of Nuclear Regulation) from the United Kingdom to broaden cooperation with the United States in areas related to the fuel cycle, storage, disposal, decommissioning, and activities associated with the post-Fukushima event for fuel cycle facilities. The insights and lessons learned gathered through interactions with the international community were used during the NRC staff evaluation of NRC regulated fuel cycle facilities.

IV. Conclusions and Recommendations

Because of the systematic and methodical evaluation of fuel facilities and GTCM licensees in light of the lessons learned from Fukushima, NRC staff concludes that the current regulatory approach and requirements for fuel cycle licensees offers reasonable assurance of adequate protection of public health and safety. For these facilities, compliance with the current regulatory framework ensures that the consequences of any potential natural phenomena event to offsite receptors and workers are within the dose limits stipulated by the regulatory requirements. NRC staff will continue its efforts to resolve concerns with fuel facilities' safety assessments and the supporting documentation with respect to the treatment of NPHs. NRC staff expects to issue a generic letter on this subject in 2015.

3. Radioactive Material Users

I. Current Regulatory Framework

The NRC regulates about 2,900 research, medical, industrial, government, and academic materials licensees. In addition, the NRC has agreements with 37 States, under which the States assume regulatory responsibility for the use of certain radioactive materials. These Agreement States oversee about 18,000 licensees. The quantities that medical and academic licensees possess can range from millicurie quantities of radionuclides to thousands of curies contained in self-shielded irradiators and gamma knife devices. Industrial uses of sealed source devices include a variety of applications and devices that include density gauges, thickness gauges, prompt gamma neutron activation analysis gauges, well logging gauges, moisture density gauges, industrial radiography sources, irradiators, as well as others.

Safety evaluations for sources and devices used in industrial, academic and medical settings are mostly evaluated as part of the sealed source and device registration process. Licensees are authorized to possess and use only those sealed sources and devices specifically approved or registered by NRC or an Agreement State. The NRC or Agreement State evaluate the safety of gauges, radiography source assemblies, exposure devices, source changers, and well-logging sources before authorizing a manufacturer or distributor to distribute the gauges to specific licensees. The safety evaluation is documented in a sealed source and device registration certificate.

Sealed sources are required to satisfy rigorous design and performance criteria. To prevent accidental dispersion from the device into the environment, the licensed material should be a chemical and physical form that is as insoluble and non-dispersible as practical. The material is doubly encapsulated; the capsule should be resistant to extreme changes in temperature, pressure, and vibration; and it is resistant to impact and puncture. The evaluation of the sources and devices include a review of the design, manufacturing, prototype testing, and proposed uses. The sources and devices are designed to survive normal conditions of use and likely accidental conditions.

Self-shielded irradiators (e.g., blood irradiators) incorporate many engineering features to protect individuals from unnecessary radiation exposure. Many other devices (e.g., gamma knife, radiography cameras, gauges, etc.) have some engineering features (shielding, connectors, switches, etc.) that prevent unnecessary exposure. These devices are usually designed for use in an industrial or laboratory environment, i.e., inside a building, protected from the weather, and in many cases without wide variations in temperature and humidity.

Facilities and equipment must be adequate to protect health, minimize danger to life or property, minimize the possibility of contamination, and keep exposure to occupationally exposed workers and the public as low as is reasonably achievable (ALARA). Licensed materials located in an unrestricted area and not in storage must be under the constant surveillance and immediate control of the licensee. Areas where material is used or stored, including below ground bunker storage areas, should (1) be accessible only by authorized persons, and (2) secured or locked when an authorized person is not physically present.

NRC regulations applicable to devices containing byproduct material and to NRC radiation safety evaluations are found in the following:

- 10 CFR Part 20, “Standards for protection against radiation”
- 10 CFR Part 21, “Reporting of defects and noncompliance”
- 10 CFR Part 30, “Rules of general applicability to domestic licensing of byproduct material”
- 10 CFR Part 31, “General domestic licenses for byproduct material”

- 10 CFR Part 32, “Specific domestic licenses to manufacture or transfer certain items containing byproduct material”
- 10 CFR Part 33, “Specific domestic licenses of broad scope for byproduct material”
- 10 CFR Part 34, “Licenses for industrial radiography and radiation safety requirements for industrial radiographic operations”
- 10 CFR Part 35, “Medical use of byproduct material”
- 10 CFR Part 37, “Physical protection of category 1 and category 2 quantities of radioactive material”[23]
- 10 CFR Part 40, “Domestic licensing of source material”
- 10 CFR Part 70, “Domestic licensing of special nuclear material” (for pacemaker devices)
- 10 CFR Part 71, “Packaging and transportation of radioactive material”
- 10 CFR Part 150, “Exemptions and continued regulatory authority in Agreement States and in offshore waters under section 274”

Licensees are responsible for the security and safe use of all licensed material (sealed and unsealed) from the time it arrives at their facility, during its use and storage, and until it is transferred or disposed. The licensee should be able to account for the location of all materials possessed, whether the material is located in a secured laboratory cabinet, a locked refrigerator or freezer, or in an appropriate waste container awaiting disposal. Security for category 1 and 2 sources (e.g., gamma knife, blood irradiators, etc.) is now described in 10 CFR Part 37, “Physical protection of category 1 and category 2 quantities of radioactive material.” This regulation establishes security requirements for the use and transport of the most risk-significant quantities of radioactive materials.

[23] Category 1 sources and practices - personally extremely dangerous: This amount of radioactive material, if not safely managed or securely protected, would be likely to cause permanent injury to a person who handled it, or were otherwise in contact with it, for more than a few minutes. It would probably be fatal to be close to this amount of unshielded material for a period of a few minutes to an hour (see IAEA-TECDOC-1344, Categorization of radioactive sources, 2003 for more details).

Category 2 sources and practices - personally very dangerous: This amount of radioactive material, if not safely managed or securely protected, could cause permanent injury to a person who handled it, or were otherwise in contact with it, for a short time (minutes to hours). It could possibly be fatal to be close to this amount of unshielded radioactive material for a period of hours to days.

Licensing

The NRC's NUREG-1556 technical report series, "Consolidated Guidance about Materials Licenses," offers a comprehensive source of reference information about various aspects of materials licensing and materials program implementation. These reports, where applicable, describe a risk-informed, performance-based approach to licensing consistent with the current regulations. Specific guidance is found in the following:

Volume 1 - Program-Specific Guidance About Portable Gauge Licenses

Volume 2 - Program-Specific Guidance About Industrial Radiography Licenses

Volume 3 - Applications for Sealed Source and Device Evaluation and Registration

Volume 4 - Program-Specific Guidance About Fixed Gauge Licenses

Volume 5 - Program-Specific Guidance About Self-Shielded Irradiator Licenses

Volume 7 - Program-Specific Guidance about Academic, Research and Development, and Other Licenses of Limited Scope Including Gas Chromatographs and X-Ray Fluorescence Analyzers

Volume 11 - Program-Specific Guidance about Licenses of Broad Scope

Volume 14 - Program-Specific Guidance About Well Logging, Tracer, and Field Flood Study Licenses

Most of these volumes have been updated and published in the *Federal Register* for public comment. The rest of the volumes are being updated and will be published in the *Federal Register* for public comment by the end of fiscal year 2016. As applicable, additional guidance on source security and safety culture have been included.

Possession limits must cover the total anticipated inventory, including licensed material in storage and waste. If the type, form, and amounts of any of the materials possessed exceed those for Category 1 and Category 2 sources (e.g., cesium-137 or cobalt-60), they must be reported to, and tracked in, the National Source Tracking System (NSTS) in accordance with 10 CFR 20.2207, "Reports of transactions involving nationally tracked sources." Such sources also may have other requirements for security of these materials under 10 CFR Part 37, "Physical protection of category 1 and category 2 quantities of radioactive material." Refer to Appendix E, "Nationally Tracked Source Thresholds," to 10 CFR Part 20, "Standards for protection against radiation," for a list of radionuclides of interest and Category 1 and 2 quantities. This regulation establishes security requirements for the use and transport of the most risk-significant quantities of radioactive materials.

Gamma well-logging sources often contain Category 2 amounts of cesium-137 while neutron oil well-logging sources contain Category 2 amounts of americium and beryllium. Radiography exposure devices also contain Category 2 amounts of cobalt-60 or iridium-192. Academic and most medical licensees possess small quantities of unsealed source material, far less than

Category 2 quantities. However, hospitals may possess blood irradiators, teletherapy devices, gamma knives, or brachytherapy sources that contain Category 1 or 2 quantities.

Inspection and Oversight

For the fabrication and use of sealed sources and devices used by radioactive material users, the NRC ensures compliance with the regulatory requirements through both periodically scheduled and special inspections. Manual Chapter 2800, "Materials Inspection Program," establishes the inspection program for licensees authorized to possess, use, transfer, and dispose of radioactive material. Inspections are conducted in accordance with formalized procedures, specifically, Inspection Procedure No. 87125, "Materials Processor/Manufacturer Programs." The objective of the procedure is to assure that (1) licensed activities are being conducted in a manner that will protect the health and safety of workers and the general public, (2) the licensed programs are being conducted in accordance with NRC requirements, and (3) the licensee is manufacturing sources or devices in accordance with commitments made to NRC.

Emergency Preparedness

Facility design and available equipment must supply sufficient engineering controls and barriers to protect the health and safety of the public and licensee employees, keep exposures to radiation and radioactive materials ALARA, and minimize the danger to life and property from the uses of the types and quantities of radioactive materials possessed by the licensee. The licensee should have and follow emergency and operational event procedures, appropriate for the sealed or unsealed material possessed, and for natural phenomena (if required), including an earthquake, a tornado, hurricane, flooding, or other phenomena as appropriate for the geographical location of the facility.

Loss or theft of licensed material, sabotage, fires, floods, etc., can adversely affect the safety of workers and members of the public. Therefore, written procedures must be developed to minimize, as much as possible, the effect of these incidents on workers, members of the public, and the environment. Licensees who possess radioactive materials in unsealed form, on foils or plated sources, or sealed in glass in excess of the quantities in 10 CFR Part 30.72, "Schedule C—Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release," must either (1) demonstrate that the maximum dose to a person offsite due to a release of radioactive materials will not exceed 0.01 Sv (1 rem) effective dose equivalent or 0.05 Gy (5 rem) to the thyroid; or (2) develop an EP for responding to a release of radioactive material. That plans should include a facility description, means and equipment for mitigating the consequences of each type of accident, methods and equipment to assess releases of radioactive material responsibilities of licensee personnel, a description of the means to promptly notify offsite response organizations, a description of the means of restoring the facility to a safe condition, and provisions for biennial onsite exercise. The emergency preparedness plan (or contingency plan) at nuclear pharmacy manufacturing facilities such as Mallinckrodt address how the facility staff will respond to major spills, fire, tornado, and earthquake. Worst case scenario assessments assume complete release of the entire radioactive inventory of affected buildings and assess the offsite total effective dose equivalent and thyroid organ exposure to members of the public. At present, most medical and academic licensees do not possess quantities of radioactive materials that exceed the values described in 10 CFR 30.72, Schedule C, and are not required to develop an EP.

Licensees are required to develop and follow operating and emergency procedures. Each year, some gauges experience equipment failures because of corrosion caused by the harsh environmental conditions. After discovery of an equipment failure, it is important to determine if the shielding and source are intact. Procedures for routine inspection, maintenance, and operability of exposure devices, survey instruments, transport containers, and storage containers are license requirements. Additional instructions are required to minimize exposure of persons in the event of an accident as well as source retrieval instructions.

The NRC and Agreement States consider that theft or the loss of sources of larger activities may cause a substantial hazard. These licensees are required by regulation to report the theft or loss of these sources to their regulatory authority to aid in the prevention of personnel exposures. Immediate NRC notification is required if radioactive material or equipment may have caused, or threaten to cause, an exposure in excess of 5 rem TEDE or 15 rem lens dose equivalent. Lost sources have, in the past, resulted in exposures to members of the public and contamination of unrestricted areas.

II. Evaluations and Assessments Prior to Fukushima

Although most equipment that uses sealed sources is very rugged, accidents occur that can temporarily or permanently damage the equipment. If this occurs while the source is outside its shielded container, there is potential for worker and public exposure. On numerous occasions ice, blowing snow or freezing rain have prevented the retraction of a source into a radiography exposure device. Structural and vehicle fires can damage the outer casing or over pack of a radiography exposure device, but the source generally remains intact and in the shielded position. Several instances have occurred where devices have been washed overboard while on route to a job site or off a deep sea oil drilling platform; divers recovered most, but not all of these devices. In other instances, trucks or tractor trailers have run over moisture density gauges with minor damage to the storage container. In those few instances where the source was dislodged from the source holder, the source was quickly found and re-inserted into the source holder and protective shielding.

In 2006, a nuclear pharmacy reported flood damage from a storm. All radioactive material (sealed sources and radiopharmaceutical material) was present and no removable contamination was present at the pharmacy. A September 2001 tornado damaged multiple buildings at the U.S. Department of Agriculture facility. The buildings contained various forms of carbon-14, tritium, iodine-125, phosphorous-32, and sulfur-35. All fume hoods and storage refrigerators, where the radioactive material was stored, were intact and the tornado did not affect the material. No other instances involving Category 1 or 2 medical or academic sources to external events are documented in the nuclear material event database. No instances in the nuclear material event database document the loss of licensee control of Category 1 industrial, medical or academic sources due to severe weather, earthquake, or flooding.

For more than 30 years, the NRC has put improvements in place in its emergency preparedness and incident response programs especially upon reviewing lessons learned after several severe natural disasters. For example, in 2005, Hurricane Katrina struck the Gulf Coast. The Federal Emergency Management Agency (FEMA) described Hurricane Katrina as, "the single most catastrophic natural disaster in U.S. history," with estimated damage exceeding $100 billion. In preparation for and after landfall, the NRC contacted its Category 1 and Category 2 licensees to obtain information on the physical status and the security of facilities and materials in those states potentially affected by the hurricane. Coordination with the Agreement States proved successful in obtaining current information regarding the status of radioactive materials located in those states. The NRC emergency response was coordinated with other Federal agencies, such as the Centers for Disease Control and Prevention, the U.S. DOE, the Environmental Protection Agency (EPA), FEMA, and the U.S. Army Corps of Engineers. If a licensee had lost control of a radiation source, DOE's aerial monitoring system was available to search for and find any missing or misplaced radiation source. The 2005 Hurricane Season Lessons Learned Task Force proposed a series of recommendations to include a recommendation involving reporting the status of risk-significant materials. These improvements were applied to NRC responses later in the 2005 hurricane season.

Extreme external events such as earthquakes, tornadoes, hurricanes, flooding, or wildfires occur every year in the United States. Many of these events have the potential to cause the loss of licensee control of radioactive material. One tool that is available to assist the NRC and Agreement States monitor the status of radioactive material is the NSTS. The NSTS is a secure web-based database designed to document the location and status of Category 1 and Category 2 radioactive sources regulated by the NRC and the Agreement States. About 1,300 licensees began reporting their Category 1 and Category 2 source information for inclusion into the NSTS in January 2009. The tracking spans the life cycle of the source from manufacture through shipment receipt, decay, and disposal. NSTS enhances the ability of the NRC and Agreement States to inspect and investigate, communicate information to other government agencies, and verify legitimate ownership and use of nationally tracked sources.

III. Post-Fukushima Event Evaluations/Assessments

Although the source integrity of a sealed source after exposure to a natural event is very likely to be retained, loss of control of a source by the licensee is a possibility. The tsunami that struck the northeast coast of Japan destroyed thousands of buildings and created approximately 20 million tons of debris. Five million tons of this debris were swept out to sea (3.5 million tons of debris was deposited along the coast of Japan and another 1.5 million tons became floating debris in the Pacific Ocean). The Government of Japan does not believe radioactive material floats among the debris.

Since the Fukushima accident, the NRC also has developed a mapping tool to give situational awareness of the NSTS's Category 1 and 2 licensees and sources as distributed across the United States. The mapping tool was developed with help from the Federal Bureau of Investigation (FBI) to support the NRC's Operations Center for incident response. The information for the mapping tool is updated quarterly and is being shared with other agencies (FBI, DOE, National Nuclear Security Administration, and Department of Homeland Security) to support their situational awareness. Tools like this, along with NSTS, are used regularly to monitor and verify source security after natural events like earthquakes, wildfires, tornadoes, hurricanes, and flooding.

NRC staff evaluated radioactive material licensees through document review of license requirements and regulatory framework, incident reports contained in the nuclear material events database, and internal meetings with license review experts. The current assessment also reviewed various external events to determine if a failure of a sealed or unsealed source could reasonably be expected to result from the event that would be more severe than previously evaluated. The following table gives an overview of the potential outcome and overall assessment of several types of events on industrial, academic, and medical uses of radioactive material. The natural events assessed and an initial evaluation is summarized below:

Effect of External Events on Radioactive Material Users

External Event	Outcome	Assessment
Flood	Challenge to structures and vehicles in which unsealed and sealed sources are used, stored, or transported; potential loss of control	• Licensees are required to meet any city, county or state requirements/ regulations regarding building construction. • The greatest concern is the loss of licensee control of radioactive material. During the license application process, the location of all unsealed and sealed source materials must be described. For those radioactive materials with activities that exceed those for Category 1 (e.g., irradiators) and Category 2 (e.g., well logging) sources, they must be reported to and tracked in the NSTS in accordance with 10 CFR 20.2207. • Emergency plans that address natural phenomenon, including an earthquake, a tornado, flooding, or other phenomena, generally are not required by industrial licensees, unless the activity limits exceed those in 10 CFR 30.72, Schedule C. However, the licensee must develop operating and emergency procedures and the operator must demonstrate an understanding of these procedures.
Seismic	Challenge to structures in which unsealed and sealed sources are used or stored. Manufacturer must check Gamma knife equipment before patient use.	
High Wind and Missiles	Challenge to structures and vehicles in which unsealed and sealed sources are used, stored, or transported; potential loss of control	
Lightning	Challenge to structures in which unsealed and sealed sources are used, stored, or transported	
Snow and Ice Loads	Challenge to structures and vehicles in which unsealed and sealed sources are used, stored, or transported	
Drought	None	
Temperature Extremes	Failure to retract radiography source because of ice formation.	
External Fire	Challenge to structures and vehicles in which unsealed and sealed sources are used, stored, or transported. Potential damage to shielding material (e.g., lead) upon exposure to extreme heat.	
Loss of Power	Some gauges and devices	

External Event	Outcome	Assessment
	would not be affected. Challenge to transport of sealed and unsealed source material. Facility control, monitoring and security systems would be inoperable once battery backup power is exhausted	

NRC staff considered the 12 NTTF recommendations for applicability to radioactive material users and evaluated to determine if any future action is warranted. The following table summarizes the result of the review.

Near-Term Task Force Recommendations and Future Actions

Recommendations		Review Result
1	Establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.
2	The task force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection of SSCs.	No Action Needed. Greatest concern is loss of licensee control. NSTS requires tracking of sources. NRC has upgraded its capability to provide the status of risk-significant materials in response to extreme events.
3	The task force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	No Action Needed. See item 2 above.
4	The task force recommends that the NRC strengthen SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events.	No Action Needed. See item 2 above. Some licensees require power to ensure facility controls for monitoring and security. For these facilities, processes and procedures are in place to ensure material is secured in the event of a loss of power.
5	The task force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.

Recommendations		Review Result
6	The task force recommends, as part of the longer term review, that the NRC identify insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-ichi accident.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.
7	The task force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	Not Applicable to these licensees.
8	The task force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs.	No Action Needed. NRC upgraded its capabilities to provide situational awareness of the NSTS's Category 1 and 2 sources. As discussed above extreme natural phenomena events have occurred that have affected this class of licensees. NRC staff continually assesses the need to improve emergency response capabilities as a result of lessons learned from these events.
9	The task force recommends that the NRC require that facility EPs address prolonged SBO and multiunit events.	No Action Needed. See item 8 above.
10	The task force recommends, as part of the longer term review, that the NRC should pursue additional emergency plan (EP) topics related to multiunit events and prolonged SBO.	No Action Needed. See item 8 above.
11	The task force recommends, as part of the longer term review, that the NRC should pursue EP topics related to decision-making, radiation monitoring, and public education.	No Action Needed. See item 8 above.
12	The task force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the reactor oversight process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.

IV. International Experience

NRC staff, in concert with the IAEA secretariat and other Member States, support an IAEA Action Plan on Nuclear Safety that includes an action entitled, "Review and Strengthen IAEA Safety Standards and Improve Their Implementation." This action calls on the Commission on Safety Standards and its' Safety Standards Committees to review and revise, as necessary, relevant IAEA safety standards. The Radiation Safety Standards Committee (RASSC) has participated in the extensive program of work undertaken by the IAEA to review all of its safety standards following the Fukushima accident. This program focused initially on the review and revision of safety requirements, followed by the supporting safety guides. This topic was first discussed by RASSC in June 2011 and is a standard agenda item at all subsequent RASSC meetings.

NRC staff has participated in a series of IAEA-sponsored international conferences. For example, the 2nd International Conference on Occupation Radiation Protection was convened to provide an opportunity for experts to discuss the various radiation protection issues, that have been highlighted by the Fukushima accident and to consider how these should be addressed at both the national and international levels. The conference agenda covered the wide range of occupational exposure situations associated with fuel cycle, reactor, industrial, and medical, activities, and from occupational exposure situations associated with naturally occurring radioactive materials. Recommendations from these conferences will be included in a broader IAEA post-Fukushima accident review that should be finalized in 2015.

V. Conclusion and Recommendations

NRC staff concludes that unsealed radioactive materials and sealed sources and devices used in industry, academia, and medicine are appropriately licensed and have sufficient engineering controls to protect the health and safety of workers and members of the public. Worker exposures are kept as low as is reasonably achievable and minimize the danger to life and property. The safety evaluation as documented in a sealed source and device registration certificate conducted before authorizing a manufacturer or distributor to distribute the radioactive sources to specific licensee assures the integrity of the device. Thousands of industrial sources have been exposed to harsh environmental stressors and licensees have developed operational and emergency procedures to deal with unplanned accidents and emergencies. Databases and mapping tools designed to track Category 1 and Category 2 radioactive sources are continually being improved and refined. Finally, specific guidance for licensing radioactive material has been updated and published for public comment. Therefore, no further study or regulatory action is warranted.

4. Part 36 Irradiators

I. Current Regulatory Framework

Gamma radiation is routinely used in industry to eliminate disease-causing insects and micro-organisms such as E. coli and salmonella. Food products, food containers, spices, fruits, plants and medical supplies are the products most commonly irradiated. To be effective, radiation exposure upward of hundreds to thousands of gray is required to sterilize these products. The process does not leave a radioactive residue or cause the treated products to become radioactive.

A Part 36 irradiator is defined as a facility that uses radioactive sealed sources for the irradiation of objects or materials and in which radiation dose rates exceeding 5 gray (500 rads) per hour exist at 1 meter from the sealed radioactive sources, but does not include irradiators in which both the sealed source and the area subject to irradiation are contained within a device (self-shielded) and are not accessible to personnel such as a blood/tissue irradiator. Part 36 irradiators generally contain Category 1 sources of cobalt-60 with activities that range from 9 PBq (250 KCi) to 1 EBq (30 MCi).

The NRC and Agreement States license several types of irradiators. Underwater irradiators are devices in which sealed sources always remain underwater and workers do not have direct access to the sources or the area subject to irradiation without entering the pool. Panoramic irradiators include those facilities where the irradiations occur in air and workers potentially could have access to the source while it is in operation. If the source is stored in a pool of water when not in use, the facility is called a panoramic wet-source-storage irradiator. The United States has over 50 wet-source-storage irradiators in use, which are licensed to use between 37 to 520 PBq (1 to 14 MCi) of cobalt-60. If a Part 36 irradiator's sources are not stored in a pool of water when not in use, it is referred to as a dry-source-storage irradiator. In this case, the sources are stored in a large shielded container. Beam type dry-source storage irradiators emit a narrow beam of radiation which irradiates the target material in air in areas potentially accessible to workers.

In a typical irradiation system, products are loaded into irradiation containers that are conveyed through a maze into a concrete or metal shielded room. Inside the shielded room, the products are exposed to gamma rays coming from a rack of cobalt-60 sources. The concrete or metal shield is designed to limit worker and public exposure to no more than 2 mrem (20 μSv) per hour at 1 meter from the shield.

Radiation safety requirements for irradiators are described in 10 CFR Part 36, "Licenses and radiation safety requirements for irradiators." This part describes the requirements for the design, construction and safe operation of irradiators under normal and emergency operating conditions. More on these issues will be described below. Security requirements for irradiators are described in 10 CFR Part 37, "Physical protection of category 1 and 2 quantities of radioactive material." This regulation establishes security requirements for the safe use and transport of the most risk-significant quantities of radioactive materials.

NRC regulations applicable to devices containing byproduct material and to NRC radiation safety evaluations are found in the following:

- 10 CFR Part 20, "Standards for protection against radiation"

- 10 CFR Part 21, "Reporting of defects and noncompliance"

- 10 CFR Part 30, “Rules of general applicability to domestic licensing of byproduct material”

- 10 CFR Part 32, “Specific domestic licenses to manufacture or transfer certain items containing byproduct material”

- 10 CFR Part 36, “Licenses and radiation safety requirements for irradiators”

- 10 CFR Part 37, “Physical protection of category 1 and category 2 quantities of radioactive materials”

- 10 CFR Part 71, “Packaging and transportation of radioactive material”

Licensing

The NRC’s NUREG-1556 technical report series, “Consolidated Guidance about Materials Licenses,” offers a comprehensive source of reference information about various aspects of materials licensing and materials program implementation. These reports, where applicable, describe a risk-informed, performance-based approach to licensing consistent with the current regulations. Specific guidance applicable to industrial irradiators is described in NUREG-1556 Volume 6, “Program-Specific Guidance about 10 CFR Part 36 Irradiator Licenses.” This NUREG volume is being updated from the previous version to include safety culture, security of radioactive materials, protection of sensitive information, and clarification of regulatory policies and guidance. The draft revision should be finalized in 2015.

Facility design and available equipment must have sufficient engineering controls and barriers to protect the health and safety of workers and members of the public, keep exposures to radiation and radioactive materials as low as is reasonably achievable, and minimize the danger to life and property from the uses of the types and quantities of radioactive materials possessed by the licensee. The licensee must have and follow emergency or abnormal event procedures including external events such as an earthquake, a tornado, flooding, or other phenomena as appropriate for the facility’s geographical location. The design of the facility should consider abnormal events such as water loss or leakage from the source storage pool, a prolonged loss of electrical power, or a fire alarm or explosion in the radiation room.

For example, during the 1980s, there was concern that source material used in irradiators might be accidentally released into the environment. With these concerns in mind, the radioactive material used in an irradiator must be as nondispersible and as insoluble as practical if the source is used in a wet-source-storage or wet-source pool irradiator. As a result, colbalt-60, a metal, is used as a source instead of cesium-137 (a soluble chloride salt). Furthermore, the source material must be doubly encapsulated in a material that is corrosion resistant. Licensees, during the application process, must supply the manufacturer’s (or distributor’s) name and model number for each requested sealed source and device. Licensees will be authorized to possess and use only those sealed sources and devices specifically approved or registered by the NRC or an Agreement State.

In most instances, steel reinforced concrete walls and ceilings are built to enclose these irradiators to ensure that public exposure is no greater than 2 mrem per hour when measured 1 meter from the shield surface. For panoramic irradiators built in seismic areas such as California, the licensee must design the reinforced concrete radiation shields to retain their integrity in the event of an earthquake by designing to the seismic requirements of an appropriate building code such as American Concrete Institute (ACI) Standard ACI 318-89, "Building Code Requirements for Reinforced Concrete," Chapter 21, "Special Provisions for Seismic Design," or local building codes, if current. The foundation of the facility should be designed with consideration given to soil characteristics to ensure that it is adequate to support the weight of the facility shield walls. Soil failure could result in shifting of, and possibly damage to, the irradiator shield. For pool irradiators, the pool should be designed to assure that it is leak resistant and the pool walls are strong enough to bear the weight of the pool water and shipping casks. The pool is also designed to ensure that a dropped cask would not fall onto the irradiator sources, and the metal components are corrosion resistant.

Inspection and Oversight

The NRC assures compliance with the regulatory requirements through both periodically scheduled and special inspections. Inspections of irradiators licensed under 10 CFR Part 36 are conducted in accordance with formalized procedures, specifically Inspection Procedure No. 81722, "Irradiator Programs." The objective of this procedure is to determine if licensed activities are being conducted in a manner that will protect the health and safety of workers and the general public. The procedures are also used to determine whether the licensed programs are being conducted in accordance with NRC requirements. The NRC also uses Inspection Manual Chapter 2815, "Construction and Preoperational Inspection of Panoramic, Wet-Source-Storage Gamma Irradiators," to determine if panoramic wet-source-storage gamma irradiators are constructed, equipped and operated in accordance with the license application.

Emergency Preparedness

The licensee is required to develop and implement emergency and abnormal event procedures for its facility. These event procedures generally address: (1) low or high water level indicator readings, an abnormal water loss, or leakage from the source storage pool; (2) a prolonged loss of electrical power; (3) a fire alarm or explosion in the radiation room; (4) interlock failure; and (5) natural phenomena, including an earthquake, tornado, flooding, or other phenomenon as appropriate for the facility's geographical location.

In addition to emergency procedures, the irradiator facility is designed and equipped to respond to numerous emergencies. For example, if electrical power at any panoramic irradiator is lost for longer than 10 seconds, the sources must automatically return to their shielded position. One facility is equipped with backup electrical generators with 5 to 7 days fuel supply because of hurricane-related frequent extended power outages. In the event of a fire, the radiation room at a panoramic irradiator must be equipped with a fire extinguishing system capable of extinguishing a fire without the entry of personnel into the room.

As part of the licensee's emergency or abnormal operating procedures, NRC notification is required when sources are stuck in an unshielded position; any fire or explosion occurs in the radiation room; there is damage to the source racks; there is structural damage to the pool liner or wall; or abnormal water loss or leakage occurs from the source storage pool. Reports to the NRC must include a telephone report within 24 hours and a written report within 30 days.

There have been several instances where source racks did not return to a shielded position. In one instance loss of electric power damaged a programmable logic control to an irradiator causing the failure of the source racks to return to a shielded position. The source racks were safely lowered mechanically. Two serious radiation-related injuries occurred at irradiator facilities during the 1970s. In both cases, the overexposures occurred when operators walked into the exposure room when the source was unshielded. The overexposures occurred because safety systems were intentionally bypassed or operating procedures were not followed. Two deaths unrelated to radiation exposure occurred at U.S irradiators. These workers were crushed while moving materials to be irradiated on a conveyor.

II. Evaluations and Assessments Prior to Fukushima

Irradiators have been safely used to sterilize food, spices, and medical supplies since the mid-1960s. First published in 1993, 10 CFR Part 36 was based, in part, on recommendations contained in American National Standards Institute standard N43.10, “Safe Design and Use of Panoramic, Wet Source Storage Gamma Irradiators.” Each license application review includes a review of the design, manufacturing, prototype testing, and proposed uses of the sources and the irradiator. Irradiators contain certain safety features. Source racks are designed to return to a safe or shielded position in the event of a loss of power, air pressure, or other external factors. Irradiators must have access controls to prevent inadvertent entry of personnel if the sources are not in the shielded position. The sources and devices are designed to survive normal conditions of use and likely accidental conditions. Most irradiators licensed for use in the United States have seismic detectors. When this detector senses excessive vibration, the source rack is automatically lowered into the pool.

III. Post-Fukushima Event Evaluations/Assessment

Since 2013, security requirements for irradiators are described in 10 CFR Part 37. This regulation describes a performance based program of security measures required for licensees that possess risk significant quantities of materials. If there is an actual, attempted, or suspicious activity related to the possible theft and diversion or radiological sabotage of risk-significant material, the licensee is required to notify both local law enforcement and the NRC immediately.

The current assessment reviewed the generic facility design and available equipment to determine if engineering controls and barriers are sufficient to protect the health and safety of the public and irradiator workers in the event of severe natural phenomena such as an earthquake or tsunami.

In the Unities States, 55 panoramic, wet-source-storage irradiators and one underwater pool irradiator, each containing a minimum of 37 PBq (1 MCi) of cobalt-60, are licensed for use. Nine of these irradiators are in seismic areas. The NRC defines a seismic area as any location where the probability of a horizontal acceleration in rock of more than 0.3 times the acceleration of gravity in 250 years is greater than 10 percent. Although not required, each of these nine irradiators is equipped with a seismic detector that causes the radiation source holder to become fully shielded automatically in the source storage pool should the seismic detector be actuated. When returned to the storage pool, the radiation reading at the pool surface should be less than 2 mrem per hour.

The average source storage pool holds between 16,700 and 18,100 gallons of water. The water level is typically maintained 3 meters (10 feet) above the top of the source material when the source rack is in the fully shielded position. The pool liner is fabricated using 3/16-inch stainless steel, encased in high density, steel reinforced concrete, generally 40 to 50 cm (15 to 20 inches) thick. The pool is specifically designed without a tie-in to the main shield so that it is free to move independently of the shield in the event of an earthquake. Water chilling systems maintain pool water temperatures between 18 degrees to 27 degrees Celsius (65 degrees to 80 degrees Fahrenheit). The water chilling system is necessary because a single 370 TBq (10,000 Ci) cobalt-60 sealed source has an estimated surface temperature of 130 degrees Celsius (265 degrees Fahrenheit). During normal operations several hundred individual sources are loaded into the source rack. Without chilling capacity, the pool water temperature will begin to elevate and level off after 5 to 7 days at approximately 65 degrees Celsius (150 degree Fahrenheit). The pool water evaporation rate will be slow, such that it would take weeks before the source material is exposed to the air. Water can be manually added to the pool. For example, two wet storage irradiators can be quickly refilled by the fire department through a designed connection point located outside each shield. It is important to note that there is no adverse effect to the source material resulting from the pool being empty. Each source is safety tested to withstand temperatures in excess of 800 degrees Celsius (1,470 degrees Fahrenheit) for 1 hour, which simulates the temperature of many hydrocarbon fires.

In the event the source rack does not return to the shielded position, the radiation shield surrounding the source is extremely robust. The walls of the shield are designed such that the radiation dose rate in areas that are normally occupied during operation of a panoramic irradiation may not exceed 2 millirem per hour at any location 30 cm or more from the wall or ceiling of the room. To achieve this level of protection, the concrete wall and ceiling thickness generally exceeds 165 to180 cm (65 to 70 inches). The design of the walls and roof is required to comply with the structural requirements of the local building code or the American Concrete Institute Standard 318-89; hence, the concrete density, rebar size, the rebar cross spacing and the number of rows is appropriate for use in a seismic area. Assuming a minimum concrete density of 147 pounds/cubic foot, the approximate weight of each radiation shield structure is at least 2,000 tons. An NRC staff review demonstrates that the structural components of Part 36 irradiators have considerable seismic capacity and will maintain their structural integrity under quite severe ground motion conditions (greater than several g's). However, during the review of the regulatory framework, the staff identified several equivalent methods for radiation shield construction as described in ANSI N43.10, "Safe Design and Use of Panoramic, Wet Source Storage Gamma Irradiators (Category IV) and Dry Source Storage Gamma Irradiators (Category II)." The staff will consider future rulemaking activities to include these equivalent methods.

In the event of a significant earthquake, the warehouse roofing and walls may collapse restricting access to the radiation control room and primary building power may be lost for an extended period of time. However, the shield is designed to retain its integrity and to ensure that no radioactive material will be released and that there will not be any off site radiation exposure.

The NRC previously evaluated the potential impact of an earthquake generated tsunami to a pool irradiator built in Honolulu, Hawaii. NRC staff considers this evaluation to be generally bounding for other irradiators. This is because, other than the irradiator located in Hawaii, the licensed irradiator facilities are either located inland or in locations that would not generate tsunamis to the extent possible in Hawaii. For the evaluation of the Hawaii irradiator, at shore, tsunami waves up to 10 m (33 ft) can reach velocities of up to 13 m/s (29 mph). Given the weight of a single sealed source and the shear velocity needed to lift a source out of the storage pool, the wave velocities associated with the largest historical tsunami would be insufficient to remove a source from the bottom of the irradiator pool even if the facility had sustained enough damage that the source rack had been destroyed. In addition, the wave velocity of a wind-generated (i.e., hurricane) storm surge is less than that associated with a tsunami. Therefore, NRC staff concludes that it is unlikely that a tsunami or hurricane will result in a loss of control of radioactive material from an industrial irradiator that would have an adverse effect on public health and safety or the environment.

During the license application process, the location of all Category 1 and Category 2 sources must be reported to and tracked in the NSTS, which the NRC maintains. The NSTS is a secure web-based database designed to document the location and status of Category 1 and Category 2 sources. This information can be used to give situational awareness in the event of a severe weather event.

NRC staff evaluated panoramic irradiator licensees through document review of license requirements and regulatory framework, incident reports contained in the nuclear material events database and the daily event notification report, and internal meetings with license review experts. The current assessment also reviewed various external events to determine if a failure of a sealed or unsealed source reasonably could be expected to result from the event that would be more severe than previously evaluated. The following table gives an overview of the potential outcome and overall assessment of several types of events on academic and medical uses of radioactive material. The natural events assessed and an initial evaluation is summarized below:

Effect of External Events on Part 36 Irradiators.

External Event	**Outcome**	**Assessment**
Flood	Challenge to external structures in which irradiator sources are used. No loss of source control.	• Warehouse roofing and walls may be damaged thus restricting access to the irradiator control room and access door until debris is removed. • The concrete/steel reinforced radiation shield, 70 to 74 inches thick, is designed to retain its integrity in the event of an earthquake by designing to the
Seismic	Challenge to external structures in which irradiator sources are used. Radiation shield has considerable seismic capacity and will maintain structural integrity under quite severe (several g's) ground acceleration.	

External Event	Outcome	Assessment
High Wind and Missiles	Challenge to external structures in which irradiator sources are used. No damage to shield integrity.	seismic requirements of an appropriate source such as ACI Standard ACI 318-89. • A radiation shield constructed with steel, 12 to 14 inches thick, is designed to retain its integrity in the event of a severe earthquake. • Emergency or abnormal event procedures must address natural phenomenon, including an earthquake, a tornado, flooding, or other phenomena, as appropriate for the facility's geographical location.
Lightning	Challenge to external structures in which irradiator sources are used.	
Snow and Ice Loads	Challenge to external structures in which irradiator sources are used. No damage to shield integrity.	
Drought	none	
Temperature Extremes	none	
External Fire	Challenge to external structures in which irradiator sources are used. No damage to sealed sources or shield integrity..	
Loss of Power	Source automatically returns to shielded position. Malfunction of source rack possible, thus requiring mechanical intervention to return sources to shielded position. Control system door interlocks failsafe closed and locked in the event of a power loss. All facility control, monitoring, and security systems would be inoperable once battery backup power is exhausted, estimated to be within 24 hours.	

NRC staff considered the 12 NTTF recommendations for applicability to Part 36 irradiators to determine if any future action is warranted. The following table summarizes the result of the review.

Near-Term Task Force Recommendations and Future Actions

Recommendations		Review Result
1	Establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment. However, during the review of the regulatory

	Recommendations	Review Result
	balances defense-in-depth and risk considerations.	framework, the staff identified several equivalent methods for radiation shield construction as described in ANSI N43.10, “Safe Design and Use of Panoramic, Wet Source Storage Gamma Irradiators (Category IV) and Dry Source Storage Gamma Irradiators (Category II).” The staff will consider future rulemaking activities to include these equivalent methods.
2	The task force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection of SSCs.	No Action Needed. The concrete/steel reinforced radiation shield, 70 to 74 inches thick, is designed to retain its integrity in the event of an earthquake by designing to the seismic requirements of an appropriate source, such as ACI Standard ACI 318-89. A radiation shield constructed with steel, 12 to 14 inches thick, is designed to retain its integrity in the event of a severe earthquake. Staff assessment determined it is unlikely that a tsunami or hurricane will result in a loss of control of radioactive material from an industrial irradiator that would have an adverse effect on public health and safety or the environment.
3	The task force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	No Action Needed. See item 2 above. For wet sources, sources are stored in pools of water. In addition each sealed source is safety tested to withstand temperatures in excess of 800 degrees Celsius for 1 hour, which simulates the temperature of many hydrocarbon fires.
4	The task force recommends that the NRC strengthen SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events.	No Action Needed. See items 2 and 3 above.
5	The task force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.
6	The task force recommends, as part of the longer term review, that the NRC identify insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.

Recommendations		Review Result
	through further study of the Fukushima Dai-ichi accident.	
7	The task force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	No Action Needed. Pool water evaporation rate will be slow, and take weeks before the source is exposed to air. Water can be manually added to the pool. It is important to note that there is no adverse effect to the source material resulting from the irradiator pool being empty.
8	The task force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs.	No Action Needed. Emergency or abnormal event procedures must address natural phenomenon, including an earthquake, a tornado, flooding or other phenomena, as appropriate for the facility's geographical location. Because of the robust nature of the source material offsite consequences are unlikely due to extreme events.
9	The task force recommends that the NRC require that facility EP address prolonged SBO and multiunit events.	No Action Needed. See item 8 above.
10	The task force recommends, as part of the longer term review, that the NRC should pursue additional EP topics related to multiunit events and prolonged SBO.	No Action Needed. See item 8 above.
11	The task force recommends, as part of the longer term review, that the NRC should pursue EP topics related to decision-making, radiation monitoring, and public education.	No Action Needed. See item 8 above.
12	The task force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the reactor oversight process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.

IV. Conclusion and Recommendations

NRC staff concludes that Part 36 irradiators are licensed appropriately and have sufficient engineering controls to protect the health and safety of workers and members of the public. Worker exposures are kept as low as is reasonably achievable and minimize the danger to life and property. Based on the available information, it is unlikely that natural phenomena (e.g., tsunamis, earthquakes, hurricanes, or fire) will result in a loss of control of radioactive material from an industrial irradiator that would have an adverse effect on public health and safety or the environment. Program specific guidance for irradiator licensing is being revised to include additional guidance on seismic design requirements for steel radiation shields, the security of radioactive materials, and clarification of regulatory policies and practices. As such, NRC staff concludes that irradiators are licensed as appropriate to the scope and potential hazard created. No further study or regulatory action for these facilities is warranted.

5. Low-Level Waste Disposal Facilities

I. Current Regulatory Framework

Low-level waste (LLW) includes items that have become contaminated with radioactive material or have become radioactive through exposure to neutron radiation. LLW is not high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in paragraphs (2), (3), and (4) of the definition of byproduct material set forth in 10 CFR 20.1003, "Definitions." LLW typically consists of contaminated protective shoe-covers and clothing, wiping rags, mops, filters, reactor water treatment residues, equipment and tools, luminous dials, medical tubes, swabs, injection needles, syringes, and laboratory animal carcasses and tissues. The radioactivity can range from just above background levels found in nature to levels found in parts that are inside the reactor vessel in a nuclear power plant. Another type of LLW, with large volumes waiting for disposal, is depleted uranium. Low-level waste is typically stored on-site by licensees, either until the radioactivity has decayed away and the affected waste can be disposed of as ordinary trash, or until amounts are large enough for shipment to a LLW disposal site in containers approved by the U.S. Department of Transportation.

LLW disposal occurs at commercially operated LLW disposal facilities that must be licensed by either the NRC or an Agreement State. The Low-level Radioactive Waste Policy Amendments Act of 1985 (the LRWPA) gave States responsibility for the disposal of their low-level radioactive waste. The Act encouraged States to enter into compacts that would allow them to dispose of radioactive waste at a common disposal facility. Currently, four commercial disposal facilities accept low-level radioactive waste: (1) EnergySolutions in Barnwell, SC; (2) U.S. Ecology in Richland, WA; (3) EnergySolutions in Clive, UT; and (4) Waste Control Specialists, LLC, in Andrews, TX. The EnergySolutions site in Barnwell is licensed by South Carolina to receive wastes in Classes A, B, and C. The facility accepts waste from Connecticut, New Jersey, and South Carolina. The State of Washington licensed the U.S. Ecology site to receive wastes in Classes A, B, and C. It accepts waste from States that belong to the Northwest Compact (Washington, Alaska, Hawaii, Idaho, Montana, Oregon, and Wyoming) and the Rocky Mountain Compact (Colorado, Nevada, and New Mexico). The State of Utah licenses the EnergySolutions site to accept Class A waste only. The facility accepts waste from all regions of the United States. Finally, the Texas Commission of Environmental Quality licenses the Waste Control Specialists site to receive waste in Classes A, B, and C. This site is the most recent of the four, opening in 2012, and accepts LLW from Texas and Vermont, 34 States that do not have operating facilities, and the Federal government.

Storage of LLW requires an NRC or Agreement State license. NRC or Agreement State regulations require the waste to be stored in a manner that keeps radiation doses to workers and members of the public below NRC-specified dose levels. Licensees must further reduce these doses to levels that are ALARA. Actual doses, in most cases, are a small fraction of the NRC limits.

Licensing

The NRC and the Agreement States share responsibilities to protect workers, the public, and the environment from LLW. Federal regulations in 10 CFR Part 61 and the NRC's LLW regulatory program, as well as compatible Agreement State regulations and oversight, play a key role in protecting the public and the environment. However, the licensees and operators of disposal facilities ultimately bear the primary responsibility for safely handling and using LLW materials.

Regulations and Guidance Documents

The NRC regulations for shallow land disposal, which are undergoing revision, are in 10 CFR Part 61. NRC staff has issued several guidance documents on 10 CFR Part 61 requirements, including the evaluation of disruptive phenomenon (e.g., NUREG-1623). Also, in certain cases, guidance for other NRC regulatory programs (e.g., NUREG-1757) may be adapted to the 10 CFR Part 61 analyses. Early guidance for the implementation of 10 CFR Part 61 (e.g., NUREG-1199 and NUREG-1200) was generally prescriptive. More recently, the review of site-specific performance assessments has become more performance-based, driven in part by the Commission's 1995 probabilistic risk assessment policy statement (60 FR 42622, August 16, 1995). NRC staff has developed more recent guidance (e.g., NUREG-1573 and NUREG-1854) for detailed LLW performance assessment and probabilistic risk-informed analyses using an enhanced performance-based approach. NRC staff developed the "Branch Technical Position (BTP) on Concentration Averaging and Encapsulation," on January 17, 1995, which covered, in part, the waste classification technical position. The BTP also defined a subset of concentration averaging and encapsulation practices that NRC staff finds acceptable in determining the concentrations of the radionuclides tabulated in 10 CFR 61.55, "Waste classification." The latest version of the BTP was issued on February 15, 2015 (80 FR 10165).

Emergency Preparedness

The LLW disposal facilities have emergency procedures they adhere to in their license. Each of the four sites has security and emergency preparedness conditions in their license. The regulations in 10 CFR Part 62, "Criteria and procedures for emergency access to non-Federal and regional low-level waste disposal facilities," authorize the NRC to grant emergency access to any non-Federal or regional LLW disposal facility or to any non-Federal disposal facility within a State that is not a member of a Compact. The terms and conditions upon which the Commission will grant this emergency access are found in 10 CFR Part 62 Subparts B and C. The regulations in this part apply to all persons who have been denied access to existing regional or non-Federal low-level radioactive waste disposal facilities and who submit a request to the Commission for a determination pursuant to this part. 10 CFR Part 62 applies only to the LLW that the States have the responsibility to dispose of pursuant to section 3(1)(a) of the LLRWPA.

The appropriate States and their licensees consider emergency preparedness for LLW facilities. As indicated above, risk or hazard at most LLW disposal facilities are minor. However, for those facilities that may experience severe climate conditions such as earthquakes, flooding, tornadoes, and high speed winds, emergency preparedness should be addressed and potential risks and hazards need to be evaluated. The main concern is severe land erosion or landslides that may cause radioactive waste either in storage or in waste disposal cells to be transported offsite to cause harm or damage to humans or the environment. Under such conditions, site-specific performance assessments, mitigation, waste containment, and emergency preparedness would be further developed. Under the proposed rule for the revision of 10 CFR Part 61, a safety case would be developed for LLW disposal facilities to ensure that they are meeting the performance objectives. The safety case would be updated periodically and use monitoring data to reflect actual site performance while taking extra protective measures as necessary.

External Events

External events are events that originate either off the site or within the boundaries of the site but from sources that are not directly involved in the operational states of the LLW disposal site or facility. The regulations in 10 CFR Part 61.50, "Disposal site suitability requirements for land disposal," requires site suitability for near-surface disposal. In this context, the primary emphasis in disposal site suitability is given to the isolation of wastes, and minimizing the potential for the occurrence of external events through site location. The disposal site's features must ensure that the long-term performance objectives of 10 CFR Part 61, Subpart C, are met. In addition, the disposal site should be capable of being characterized, modeled, analyzed and monitored. Within the facility's region or State, a disposal site should be selected so that projected population growth and future developments are not likely to affect the ability of the disposal facility to meet the performance objectives of 10 CFR Part 61, Subpart C. Areas having known natural resources which, if exploited, would result in failure to meet the performance objectives of 10 CFR Part 61, Subpart C, must be avoided.

The disposal site must be well drained and free of areas of flooding or frequent ponding. Waste disposal should not take place in a 100-year flood plain, coastal high-hazard area or wetland, as defined in Executive Order 11988, "Floodplain Management Guidelines." Upstream drainage areas must be minimized to decrease the amount of runoff which could erode or inundate waste disposal units. The disposal site must offer sufficient depth to the water table that ground water intrusion into the waste, perennial or otherwise, will not occur. The NRC will consider an exception to this requirement to allow disposal below the water table if it can be conclusively shown that disposal site characteristics will result in molecular diffusion being the predominant means of radionuclide movement and the rate of movement will result in the performance objectives of 10 CFR Part 61, Subpart C, being met. In no case will waste disposal be permitted in the zone of fluctuation of the water table. As stated in 10 CFR 61.50(a), the hydro-geologic unit used for disposal must not discharge ground water to the land surface within the disposal site. Areas must be avoided where tectonic processes such as faulting, folding, seismic activity, or volcanism may occur with such frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of 10 CFR Part 61, Subpart C, or may preclude defensible modeling and the prediction of long-term effects. Areas must be avoided where surface geologic processes such as mass wasting, erosion, slumping, land sliding, or weathering occur with such frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of Subpart C of this part, or may preclude defensible modeling and prediction of long-term impacts. The disposal site must not be located where nearby facilities or activities could adversely affect the ability of the site to meet the performance objectives of 10 CFR Part 61, Subpart C, or significantly mask the environmental monitoring program.

Before granting a license, all of the above site suitability features must be addressed in terms of potential hazards that may affect compliance with the LLW disposal facility performance objectives. In addition, the licensee shall follow emergency and operational event procedures which are appropriate for the LLW facilities and onsite activities (e.g., waste storage or treatment). Except for certain radionuclide releases within regulatory acceptable limits, no instances involving external events at LLW facilities have been reported. It is unlikely that events such as severe weather, earthquake, or seismic events would have significant safety effects during a compliance period of 1,000 to 2,000 years.

II. Evaluations and Assessments Prior to Fukushima

The NRC relies on the four performance objectives for 10 CFR Part 61 (i.e., protection of the public, protection of intruders, protection of workers, and site stability) to mitigate disruptive processes as well as other associated requirements that were developed, in part, based on the technical analyses of the consequences of disruptive events.

Disruptive Events and Processes

The potential for disruptive processes was recognized when the NRC developed 10 CFR Part 61 in 1982. The performance objectives and other requirements were designed to limit the effects of disruptive events on the performance of a low-level waste disposal facility. The requirements applied a defense-in-depth approach, which include:

- waste characteristics – provides limits to radiological concentrations and hazardous characteristics (e.g., flammability, pyrophoricity)
- siting characteristics – supplies required or exclusionary characteristics that limit the potential for disruptive processes and events
- waste classification system – requires higher concentration waste to be disposed of more robustly (e.g., deeper)
- system design – provides a description of the active systems that cannot be used for long-term safety
- performance objectives – provides that a system must be capable of being modeled and analyzed to demonstrate that the performance objectives will be met (performance objectives include radiological protection as well as site stability)

After waste is placed in disposal cells and those cells are closed, the waste is covered by about 3 to 10 meters or more of soil, rock, and other materials. Placement of the waste underground makes rapid release of the waste to the environment extremely unlikely. Rapid release of radioactivity to the environment from a closed LLW disposal facility requires a high energy event such as from a volcano. The limited radiological content of the waste combined with the high dispersion associated with the event results in a risk-limiting state. Disruptive events could affect the passive performance of a LLW disposal facility, if passive performance is relied upon. Because the waste is buried at moderate to significant depths below the land surface, there are substantial delays between the release time from the facility and the release time to the accessible environment. The delay times are in the range of tens to thousands of years depending on the facility, location, and relevant pathways. Even at the minimal end, adequate time is available for remedial response.

III. Post-Fukushima Event Evaluations/Assessments

In response to the Fukushima Lessons Learned activities, the NRC staff has assessed the mitigation of effects of disruptive events and processes on the disposal of LLW. The assessment involved a review of the pertinent regulatory requirements, guidance documents, and technical analyses that have been performed. The combination of historical regulatory requirements and guidance documents, technical analysis, newly proposed regulatory requirements, and revised guidance documents are sufficient to mitigate the risk from disruptive events on LLW disposal facilities. In addition, because of the inherent characteristics of the waste and disposal facility designs, LLW disposal facilities are inherently resistive to the more likely disruptive processes.

Disruptive Events Guidance Documents

NRC staff has developed new draft guidance to support the revisions to 10 CFR Part 61 and to supplement existing guidance. Chapter 2 of the draft guidance document and its associated appendices offer detailed guidance for determining the proper scope of the analysis (i.e., the Features Events and Process) to incorporate the site-specific analysis of disruptive phenomenon. NRC staff developed hazard maps of many disruptive processes to facilitate this screening and analysis process. After a licensee identifies and characterizes potential disruptive processes, they may use different technical assessment methods to understand the effect of the disruptive processes on the ability of a disposal site to meet the 10 CFR Part 61 performance objectives.

In addition to the technical analyses, licensees may use engineered barriers to mitigate the effect of disruptive processes. Based on the characteristics of low-level waste disposal facilities, NRC staff expects erosive processes to be the most likely of all of the disruptive processes to affect the long-term stability of most disposal facilities. Therefore, licensees develop robust erosion control designs using durable materials which are usually developed based on the consideration of low-probability events, such as the "Probable Maximum Precipitation (PMP) and corresponding Probable Maximum Flood (PMF)," NUREG-1623. The PMF is defined as the hypothetical flood that is considered to be the most severe flood reasonably possible. It is derived based on (1) comprehensive hydro-meteorological application of the PMP, and (2) other hydrologic factors favorable for maximum flood runoff, such as sequential storms and snowmelt. Geomorphic hazard assessment may be used to develop the erosion protection designs.

The inadvertent intruder analysis for 10 CFR 61.42 has a key role with respect to mitigating the effect of disruptive events on a low-level waste disposal facility. In 10 CFR Part 61, NRC envisioned that the inadvertent use of the disposal site may occur after closure. To ensure that a member of the public who inadvertently intrudes into the disposal site was protected, NRC developed intruder scenarios and calculated the doses that could result from exposure to a unit concentration of the isotopes anticipated to be present in future LLW disposals. A concentration that was equivalent to 500 mrem was then calculated and formed the basis for the waste classification tables. The intruder scenarios considered intrusions such as, drilling, home construction, discovery, and agriculture development. These calculations and results are described in the draft and final Environmental Impact Statements for 10 CFR Part 61, as well as an update to the analysis published in 1986 (NRC, 1981; NRC, 1982; NRC,1986). NRC staff has determined that the intruder scenarios bound the risk from other potential disruptive events. The unlikely probability of the scenarios occurring and the limited dispersion involved in the intruder scenarios combine to make the intruder scenarios more limiting. In addition to the intruder scenarios, exposed waste scenarios and operational scenarios were also assessed. As discussed above, robust erosion control covers are expected. However, the risk associated with total or partial removal of the cover was also assessed. The covered waste was assumed to be exposed and dispersed by wind or water. The transport mechanism used was the transport of radionuclides via surface water runoff leading to the exposure of individuals from a contaminated water body or wind transport. The entire disposal facility was assumed to be exposed. The surface water erosion rate was about 0.82 tons/acre-yr. Effects were calculated for time periods of up to 1,000 or 2,000 years after closure. The maximum estimated dose was about 1 mrem per year.

Operational scenarios assessed included a container accident and an accident involving fire. In the container-accident scenario, an accident is assumed to occur in which a container drops from a significant height, breaks open, and releases a portion of its contents into the air. Atmospheric dispersion was included for offsite impact calculations. The fire scenario involves a hypothetical fire in a disposal cell that lasts for two hours. The fire causes radioactivity to be released into the air. Exposures from a fire scenario are a strong function of the waste form, facility design, and operation. The maximum volume-weighted effect from a fire scenario was about 32 mrem per year.

NRC staff evaluated seismic activities as potential natural disruptive events that could affect current commercial LLW disposal facilities. In this regard, NRC staff evaluated the most recent USGS seismic activity maps, particularly recent maps showing 2 percent probability exceedance in 50 years of different peak ground acceleration distributions. Based on this evaluation, NRC staff concluded that LLW commercial disposal facilities are located in areas with peak ground acceleration in the range of 0.02 – 0.2g, where g is the standard gravity 9.8 m/s^2. The Fukushima earthquake was estimated at 2.99 g which is about 15 times, or more, stronger than any anticipated earthquake at United States commercial LLW disposal facilities. Therefore, the potential damage for typical buildings above the surface is anticipated to be light to moderate. For shallow land disposal facilities, waste is placed below the surface and covered with a thick earthen cover of soil and rocks. The potential damage to the LLW structure and the potential release of radioactive materials in this type of a facility are negligible. Thus, NRC staff's predicted seismic hazards events would cause no significant radiological effects to the public. Based on the above discussion and analysis, the doses to members of the public under extreme natural events are not expected to exceed the EPA protective action guideline levels under emergency situations. The following table gives an overview of the potential outcome and overall assessment of several types of events on LLW facilities. The natural events assessed and an initial evaluation is summarized below:

Effect of External Events on LLW Facilities

<table>
<tr><th>External Event</th><th>Outcome</th><th>Assessment</th></tr>
<tr><td>Flood</td><td>• Challenge to LLW storage and treatment facilities, as well as disposal cells before emplacement of cover.
• Challenge to onsite waste shipments waiting for disposal.
• Challenge to waste acceptance criteria for contents of liquids.
• Challenge to integrity of the disposal cell after placement of cover.</td><td rowspan="7">The three LLW disposal sites in Utah, Richland, and Texas are located in a dry environment. Therefore, severe flooding is unlikely to occur. For South Carolina, LLW disposal site flooding may cause release of LLW materials particularly from uncovered cells; however; potential hazards/risk would be anticipated to be low because of dilution and mixing with surface materials. For all LLW facilities, disposal cells are designed to allow rain drainage, thus avoiding release of materials in disposal cells. It is noted that all four LLW disposal sites are in low seismic hazards areas. In addition, because of design, and the engineered and natural barrier systems at these facilities, there should be no significant safety effects. Furthermore, waste packages and disposal operation at the facility should minimize release of radioactive materials, and thereby minimize intrusions and provide additional defense-in-depth for safety. More importantly, the current waste acceptance criteria of waste classes and disposal at the sites should offer additional assurance of low hazard or risk from such external event.

External events have been addressed in accordance with 10 CFR Part 61 requirements; therefore, EPs that address natural phenomena, including earthquake, a tornado, flooding, or other phenomena, generally should not be required for LLW disposal facilities.

For certain specific waste streams with large volumes and long-lived radionuclides (e.g., DU) more guidance and site-specific risk/hazard</td></tr>
<tr><td>Seismic</td><td>• Challenge to structures and integrity of containers.
• Challenges to vaults and covers to contain LLW and minimize intrusion.
• Challenges to integrity of disposal cells because of mechanical stresses or because of water infiltration through faults or cracks.</td></tr>
<tr><td>High Wind and Missiles</td><td>Challenge to structures in which LLW radioactive material is contained or stored in a transition before disposal.</td></tr>
<tr><td>Lightning</td><td>Challenge to structures that could catch fire during lightning which may contain stored pyrophoric radioactive materials in a transition before disposal.</td></tr>
<tr><td>Snow and Ice Loads</td><td>Challenge to structures in which radioactive material is stored or being treated for disposition.</td></tr>
<tr><td>Drought</td><td>None</td></tr>
<tr><td>Temperature Extremes</td><td>None except for pyrophoric materials stored before disposition.</td></tr>
</table>

External Event	Outcome	Assessment
External Fire	Challenge to structures in radioactive materials stored in a transition for disposition particularly pyrophoric materials.	assessments are being developed in the context of a limited rulemaking for 10 CFR Part 61. Such site-specific analysis should offer extra protective assurance particularly if climate conditions change in the long-term future.
Loss of Power	None	

NRC staff considered the 12 NTTF recommendations for applicability to LLW facilities and evaluated to determine if any future action is warranted. The following table summarizes the result of the review.

Near-Term Task Force Recommendations and Future Actions

Recommendations		Review Result
1	Establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.
2	The task force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection of SSCs.	No Action Needed. Because of the inherent characteristics of the waste and disposal facilities' designs, LLW disposal facilities are inherently resistive to disruptive events. Waste acceptance criteria provide additional assurance of a low hazard or low risk from external events.
3	The task force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	No Action Needed. See item 2 above.
4	The task force recommends that the NRC strengthen SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events.	Not Applicable. See item 2 above.
5	The task force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.

Recommendations		Review Result
6	The task force recommends, as part of the longer term review, that the NRC identifies insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-ichi accident.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.
7	The task force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	Not Applicable.
8	The task force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs.	No Action Needed. Licensees consider emergency preparedness of LLW facilities. Main concern is severe land erosion or landslides that may cause radioactive waste in storage or waste disposal cells to be transported offsite. However, risks or hazards are considered low based on the classification of the waste disposed in these facilities and design of the facilities being resistive to disruptive events.
9	The task force recommends that the NRC require that facility EPs address prolonged SBO and multiunit events.	No Action Needed. See item 8 above.
10	The task force recommends, as part of the longer term review, that the NRC should pursue additional EP topics related to multiunit events and prolonged SBO.	No Action Needed. See item 8 above.
11	The task force recommends, as part of the longer term review, that the NRC should pursue EP topics related to decision-making, radiation monitoring, and public education.	No Action Needed. See item 8 above.
12	The task force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the reactor oversight process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment. For certain specific waste streams with large volumes and long-lived radionuclides (e.g., DU) more guidance and site-specific risk/hazard assessments are being developed in the context of 10 CFR Part 61 limited rulemaking.

IV. International Experience

NRC staff are actively engaged in international activities pertaining to post-Fukushima lessons learned applicable to NRC activities in LLW management, uranium recovery (discussed below in Section 6), and decommissioning (discussed below in Section 7). The NMSS/DUWP Director is the United States representative on the IAEA Waste Safety Standards Committee (WASSC). The NMSS/DUWP Director participated in the WASSC meetings after the Fukushima accident, as well as in the "IAEA Expert Meeting on Decommissioning and Remediation after Accident," which was held at IAEA Headquarters in January 2013. NRC staff also participated in the review of IAEA standards (e.g., safety requirements), especially those modified in response to the IAEA "ACTION Plan" after the Fukushima accident. NRC staff also participated in Nuclear Energy Agency (NEA)/OECD activities particularly the "Radioactive Waste Management Committee" (RWMC). The NMSS Director is the United States representative on RWMC. The NRC staff participated in RWMC workgroup or task groups such as the Working Party on Dismantling and Decommissioning and its subsidiary task groups.

It is noted that no IAEA waste safety standards requirement were identified to be modified based on Fukushima lessons learned. In addition, throughout NRC staff and management international participation, there were no significant safety issues or actions identified that need to be undertaken by the NRC to modify its regulations for activities involving LLW management, decommissioning, and uranium recovery. In fact, in several cases, NRC regulations are more restrictive to ensure safety than IAEA safety standards, particularly those pertaining to safety under emergency situation and post remedial actions after a severe accident.

V. Conclusion and Recommendations

The combination of current regulatory requirements and guidance documents, technical analysis, and the newly proposed regulatory requirements and revised guidance documents, are sufficient to mitigate the risk from disruptive events on LLW disposal facilities. The current locations of commercial LLW disposal facilities are in low to moderate potential seismic damage regions; thus anticipated damage to disposal facilities with significant releases of radioactive materials is unlikely. No other technical analyses are necessary at this time. However, for licensing of new facilities or renewal of existing facilities, the addition of the features, events, and process analyses requirements combined with the site-specific intruder assessment and site stability analyses will ensure that disruptive events are appropriately accounted for in licensing of LLW disposal facilities that may dispose of new waste streams, such as depleted uranium. The LLW commercial disposal facilities are sited to avoid disruptive events or processes as practicable. There are no specific protective action guideline limits required for LLW similar to reactors to design barriers to mitigate potential radionuclide releases from seismic events. Nevertheless; regulatory requirements for stability of LLW disposal facilities, as evident from current locations of commercially licensed LLW disposal facilities, ensure that safety of the public and the environment are protected from such disruptive events.

6. Uranium Recovery Facilities and Uranium Mill Tailings

I. Current Regulatory Framework

Uranium recovery involves extracting natural uranium ore from the earth and concentrating (or milling) that ore. These recovery operations produce a product, called "yellowcake," which is then taken to a fuel cycle facility for further processing and enrichment. The yellowcake is transformed into fuel for nuclear power reactors. In addition to yellowcake, uranium recovery operations generate waste products, called byproduct material, which contain low levels of radioactivity. The NRC does not regulate conventional uranium mining (where ore is removed from open pits or underground shafts). The NRC regulates in situ recovery (ISR) (formerly known as in situ leach recovery) where the uranium ore is chemically altered underground before being pumped to the surface for further processing. The NRC also regulates conventional uranium milling where uranium ore is processed at the surface.

In the United States, the ISR process is the most widely used technique to extract uranium from below the ground surface. In the ISR process, injection wells pump a chemical solution into the layer of earth containing uranium ore. The solution dissolves the uranium from the deposit in the ground, and is then pumped back to the surface through recovery wells and sent to the processing plant to be converted into uranium yellowcake which is a solid form of mixed uranium oxide. Yellowcake is commonly referred to compounds such as U_3O_8, and UO_4H_2O, because these chemical compounds comprise more than 85 percent of the yellowcake produced by UR facilities. This product (i.e., yellowcake) is then transported to a uranium conversion facility, where it is transformed into uranium hexafluoride (UF_6), in preparation for fabricating fuel for nuclear reactors. Monitoring wells are installed at ISRs to ensure that uranium and chemicals are not escaping from the drilling area.

At a conventional uranium mill, uranium ore from a mine is first crushed into sand sized particles. Then a leaching agent is used to remove the uranium from the ore. In addition to extracting 90 to 95 percent of the uranium from the ore, the leaching agent also extracts several other "heavy metal" constituents, including molybdenum, vanadium, selenium, iron, lead, and arsenic. A series of other chemical processes are used to further refine, remove chemical impurities, and concentrate the uranium into yellowcake. Similar to what occurs at an ISR facility, the finished yellowcake is packed in 55-gallon drums and transported to a uranium conversion facility. The mill tailings generated during the conventional milling process are considered byproduct material. Mill tailings typically are disposed of in an impoundment and are covered with a radon barrier to minimize the amount of radon leaving the impoundment. In addition to radon emanation, mill tailings also contain other heavy metals that are typically found with uranium.

In general, the primary industrial hazards associated with uranium milling are the occupational hazards found in any metal milling operation that uses chemical extraction, as well as the chemical toxicity of the uranium itself. The main risk from uranium is intake through ingestion or inhalation. In addition, hazards of radon gas and decay progenies need to be addressed for protection of workers and the public. Because the uranium produced at these facilities is not enriched, there is no criticality hazard and little concern of fire or explosive hazards. Radiological hazards are also of negligible concern because uranium has little penetrating radiation and only moderate non-penetrating radiation. Therefore, the primary radiological hazard is attributable to the presence of radium in the byproduct material.

Licensing and Regulations

The NRC currently regulates uranium recovery operations in Wyoming, New Mexico and Nebraska. By issuing or amending a current license, the NRC authorizes the licensee to construct and operate (with specified conditions) a uranium recovery facility, expand an existing facility, or restart an existing facility at a specific site, in accordance with established laws and regulations.

Uranium mill tailings are primarily the sandy processed waste material from a conventional uranium mill. This ore residue contains the radioactive decay products from the uranium chains (mainly the U-238 chain) and heavy metals. As defined in 10 CFR Part 40*,* the tailings or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content is byproduct material. This includes discrete surface waste resulting from uranium solution extraction processes, such as in situ recovery, heap leach, and ion-exchange. Byproduct material does not include underground ore bodies depleted by solution extraction. The wastes from these solution extraction facilities are transported to a mill tailings impoundment for disposal.

The requirements for this type of byproduct material are described in 10 CFR Part 40, Appendix A, “Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the Extraction or Concentration of Source Material from Ores Processed Primarily for Their Source Material Content.” In particular, the criteria in Appendix A cover the siting and design of tailings impoundments, disposal of tailings or wastes, decommissioning of land and structures, ground water protection standards, testing of the radon emission rate from the impoundment cover, monitoring programs, airborne effluent and offsite exposure limits, inspection of retention systems, financial surety requirements for decommissioning and long-term surveillance and control of the tailings impoundment, and eventual government ownership of the tailings site under an NRC general license. Licensees and applicants may propose alternatives to the specific requirements of Appendix A for approval by the Commission.

Mill tailings could pose a hazard to public health and safety. As the radium decays over thousands of years, tailings produce a radioactive gas called radon. To keep them isolated, the tailings are placed in “tailings piles” for long-term storage or disposal. A tailings pile could be a large trench or a former mine pit and must meet NRC regulations. These piles generally are lined, covered, and monitored to detect leaks. The NRC also requires that adequate funds will be available to decommission the site, properly close the tailings pile, and maintain and monitor the site over the long term.

The U.S. Congress passed the Uranium Mill Tailings Radiation Control Act in 1978. This law created two programs to protect the public and the environment from uranium mill tailings. Among other things, this law made DOE responsible to assume control of uranium processing sites after closure for long term monitoring, although a State may voluntarily assume such control.

Title I – Legacy Sites

Many sites that produced uranium for the early nuclear power and weapons programs closed before requirements had been established for cleanup and long-term maintenance. Under the law, as of May 18, 2015, the DOE was responsible for cleanup at 22 legacy sites. All of these sites are in western states, except for two sites in Pennsylvania.

The NRC reviewed DOE's plans for cleanup and long-term maintenance of the Title I sites. The DOE has completed surface clean-up and is continuing to clean up ground water at several sites. The DOE maintains these sites and supplies long-term surveillance under a general license from the NRC.

Title II – Sites Licensed in 1978 or Later

The NRC's primary implementing regulations for these sites are found in 10 CFR Part 40. These sites must be licensed by the NRC or an Agreement State. For all uranium sites, the following apply:

- Holders of licenses to recover uranium are responsible for compliance with and the cost of site decontamination, decommissioning, and reclamation requirements issued by the NRC.

- Facility design must be submitted for review with the application.

- Plans to clean up any surface or ground water contamination must be approved by the NRC or the Agreement State in which the site is located.

For conventional mill facilities, the following apply:

- When a site has ceased operations and is being prepared for closure, DOE develops a long-term surveillance plan for the site that must be reviewed by the NRC.

- When the NRC finds a site meets the cleanup standards or concurs that a state-licensed site meets the standards, and the tailings pile meets the approved design criteria, the NRC or state can terminate the license.

- After the NRC has accepted DOE's long-term surveillance plan and terminated the license, the site transfers to DOE for long-term surveillance and maintenance under a general license.

- A State can assume responsibility for long-term care and maintenance of the site, but to date, this has not occurred.

In 1983, the EPA issued uranium mill tailings standards for both Title I and Title II sites. In 1985 and 1987, the NRC updated its regulations for Title II sites to be consistent with EPA's standards. In 1995, the EPA issued final standards for cleaning up ground water at Title I sites.

Currently there are 28 uranium mill sites that are in varying licensing status. Eleven NRC-licensed uranium mill sites closed and are being decommissioned. Nine sites licensed by Agreement States are being decommissioned. Six sites are fully decommissioned (four NRC sites and two in Agreement States) and were transferred to DOE for long-term monitoring. The NRC licensed a facility in Clive, Utah, in 1993, to take mill tailings for disposal. In 2004, Utah assumed regulatory authority for the Clive disposal site as an Agreement State. The Texas Commission for Environmental Quality (TCEQ) licensed a facility to take mill tailings for disposal near Andrews, Texas, in 2008.

A uranium recovery license is typically valid for 10 years from the date of the Commission finding under 10 CFR 40.32 based on current NRC licensing practice. A uranium recovery license can be renewed in 10-year increments. As of April 2014, the NRC is reviewing eight such actions for a new facility, expansion of an existing facility, or renewal of an existing facility. Additional applicable regulations include:

- 10 CFR Part 20, "Standards for protection against radiation"

- Appendix A to 10 CFR Part 40, "Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the Extraction or Concentration of Source Material from Ores Processed Primarily for Their Source Material Content"

Appendix A to 10 CFR Part 40 sets forth the criteria relating to the operation of uranium mills and the disposition of tailings or wastes produced by the extraction or concentration of source material from ores processed primarily for their source material content. In general, these criteria require uranium recovery facilities to control industrial hazards and address waste and decommissioning concerns.

Inspection and Oversight

NRC inspects licensed facilities and covers areas such as training of personnel who operate a uranium recovery facility, radiation protection programs, dose records, and security of nuclear materials onsite and during shipment.

The goals of the inspection program at uranium recovery facilities are:

- Ensure operations are conducted safely and in accordance with license conditions and NRC's regulations and requirements.

- Ensure, through direct observation and verification, that uranium recovery facilities are taking appropriate corrective actions to address violations of operational safety procedures.

- Ensure that a uranium recovery site is minimizing contamination and exposure to the workers and to the public using ALARA concept throughout its operation including packaging of yellowcake into safe containers as required by NRC and DOT regulations.

The NRC does not monitor radioactive releases independently on a continuous basis, but performs planned and unplanned inspections in coordination with EPA and Agreement States.

Independent measurement of the releases to verify the licensee's samples may be conducted in coordination with the concerned State.

Emergency Preparedness

Previous NRC staff assessments have evaluated the risks at both conventional uranium mills and ISR facilities. Specifically, NUREG-0706, “Final Generic Environmental Impact Statement on Uranium Milling,” considered risks at conventional uranium mills and evaluated several scenarios resulting in a release of radioactive materials. NUREG-0706 found situations where workers could receive a dose from these scenarios (but still within the Part 20 limits); the report did not find significant doses to members of the public resulting from the evaluated scenarios. NUREG/CR-6733, “A Baseline Risk-Informed, Performance-Based Approach for In Situ Leach Uranium Extraction Licensees,” presents NRC staff analyses of in situ leach uranium extraction facility operations and accidents that consider both likelihood of occurrence and the consequence. The analyses in NUREG/CR-6733 are conservative and demonstrate that in situ leach uranium extraction facilities operated with properly trained workers and effective emergency response procedures generally pose low levels of radiologic risk. NRC staff considers analyses similar to, or based on, those in NUREG/CR-6733 to be an appropriate basis for licensee safety analyses.

The guidance in NUREG-1569, “Standard Review Plan for In Situ Leach Uranium Extraction License Applications,” does not state that applicants should have complex accident analyses, consequence evaluations, and probability determinations. However, site-specific conditions and circumstances must be addressed in any uranium recovery application. The NRC has evaluated the effects of accidents at conventional uranium mills (NUREG-0706). These analyses demonstrate that, for most credible potential accidents, consequences are minor so long as effective emergency procedures and properly trained personnel are used. The previous studies found situations where workers may be subjected to higher than anticipated normal doses, but the doses would remain within the 10 CFR Part 20 limits. Doses to members of the public were expected to be lower than the worker doses.

External Events

External events are events that originate either off-site or within the boundaries of the site but from sources that are not directly involved in the operation of uranium recovery facilities. External events hazards and potential effects are appropriately evaluated during licensing or license amendment reviews. External events are important aspects of uranium recovery licensing and inspection process.

The regional location and site layout for a proposed uranium recovery facility is extensively reviewed to show the relationship of the site to local water bodies (lakes and streams); geographic features (highlands, forests); geologic features (faults, folds, outcrops); transportation links (roads, rails, airports, waterways); political subdivisions (counties, townships); population centers (cities, towns); historical and archeological features; key species habitat; and non-applicant property (farms, settlements). In this regards, maps are reviewed to examine geologic features, structures such as surface impoundments, diversion channels, monitoring wells, and recovery plant buildings.

NRC staff also considers meteorology through a description of the general climate of the region and local meteorological conditions based on appropriate data from the National Weather Service, military, or other stations recognized as standard installations. NRC staff considers potential events based on review of data on precipitation, evaporation, and joint-frequency distribution data by wind direction, wind speed, stability class, period of record, and height of data measurement. The on-site program should be designed in accordance with Regulatory Guide 3.63, "Onsite Meteorological Measurement Program for Uranium Recovery Facilities—Data Acquisition and Reporting" (NRC, 1988). The meteorological data used for assessing effects are substantiated as being representative of expected long-term conditions at and near the site.

NRC staff also considered the seismicity and the seismic history of the region. In an application, historical seismicity data should be summarized on a regional earthquake epicenter map, including magnitude, location, and date of all known seismic events. Where possible, seismic events associated with the tectonic features described in the geologic structures should be included. Flooding, faulting, folding, seismic activities, volcanism, meteorology, climate/climatology, and surface water, as well as the hydrological system at the facility must be addressed in terms of potential hazards that could affect compliance with the uranium recovery facility safety objectives.

II. Post-Fukushima Event Evaluations/Assessments

NRC staff evaluated various external events to determine if a failure of safety systems or barrier(s) at uranium recovery or mill tailing facilities could cause significant release of radioactive materials to harm workers or the public or cause any significant damage to the environment. In summary, external events such as flooding, seismic, or external fire hazards are not expected to result it doses in excess of NRC dose limits. For ISR facilities, the process of uranium extraction is conducted below ground. Therefore, potential effects from seismic or flood hazards would be negligible. For mill tailing facilities, the tailings are placed in piles at or below surface and stabilized with thick clay covers and heavy rip-rap boulders to minimize erosion and to stabilize materials under severe conditions. Therefore, effects from external events on mill tailing piles should not exceed public dose limits. Under severe external events conditions, potential dose effects to members of the public should not exceed EPA protective action guidelines under emergency situation based on an NRC staff assessment. Therefore, the potential risk for all external events is low. In summary, for licensed uranium recovery and mill tailings facilities, NRC staff has found that under extreme external events or conditions, dose impacts would not exceed regulatory criteria for members of the public under an emergency situation. The following table gives an overview of the potential outcome and overall assessment of several types of events on uranium recovery and mill tailing facilities. The natural events assessed and an initial evaluation is summarized below:

Effect of External Events on Uranium Recovery and Mill Tailing Facilities

External Event	Outcome	Assessment
Flood	• Challenge to uranium recovery structures, impoundments, storage,	External events are addressed in the license application. Therefore, severe flooding is unlikely to occur or cause

External Event	Outcome	Assessment
	and treatment facilities (e.g., ion exchange, calcination, and drying), as well as byproduct waste before disposition or permanent disposal. • Challenge to onsite yellowcake packaging and shipments waiting for transport to fuel cycle treatment and enrichment facilities. • Challenge to acceptance criteria for environmental releases and contents of liquids. • May challenge integrity of the impoundment and buried piping between the well fields and the processing facilities. • Uncontrolled chemical releases.	significant harm to uranium recovery facilities. Although severe flooding may cause low-activity releases from uranium recovery impoundments, the consequences of such a release would be low. The combination of siting and design requirements, existing drainage systems, and engineered and natural barrier systems at uranium recovery facilities are expected to prevent significant safety effects from external events. Consequences of extreme conditions (e.g., from (i) radon releases from process streams; (ii) yellowcake dryer hazards; (iii) lixiviate leaks in buried piping between the well fields and the processing facility; and (iv) chemical accidents) are expected to be less than EPA protective action guidelines.
Seismic	• Challenge to uranium recovery structures and integrity of containers. • Challenges to vaults and covers to contain uranium recovery impoundments and byproduct materials waste. • Challenges to integrity of process streams and potential radon release. • Challenges to the buried piping and infrastructure because of lixiviant leaks. • Challenges to integrity of chemical treatment system and potential environmental damage. • Challenges to integrity of yellowcake.	

External Event	Outcome	Assessment
High Wind and Missiles	Challenge to structures in which uranium recovery equipment and engineered systems are housed. Radon releases as well as low-activity uranium and chemicals may be released.	
Lightning	Challenge to structures that may catch fire during lightning strikes which may contain stored yellowcake.	
Snow and Ice Loads	Challenge to structures in which ore is being processed for concentration, calcination, and drying; and to byproduct waste in a transition before disposal.	
Drought	None	
Temperature Extremes	None in the cold regions. In the hot and dry regions concerns of fire from oxidizing material stored before shipment.	
External Fire	Challenge to structures in which uranium recovery plant and equipment are housed. Effects on uranium oxides, yellow cake, stored before shipment could be significant. Oxidizing nature of yellowcake could cause a fire hazard. Assignment of risk from external hazard due to the oxidizing nature of yellowcake is low.	
Loss of Power	Possible release of radon.	

NRC staff considered the 12 NTTF recommendations for applicability to uranium recovery and mill tailing facilities to determine if any future action is warranted. The following table summarizes the result of the review.

Near-Term Task Force Recommendations and Future Actions

Recommendations		Review Result
1	Establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.
2	The task force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection of SSCs.	No Action Needed. For ISR facilities, uranium extraction is conducted in the subsurface. For mill tailings facilities, the tailings are placed in piles at or below the surface and stabilized (e.g., use of clay and rip-rap boulders). Under severe external event conditions dose effects to members of the public are not expected to exceed EPA protective action guidelines.
3	The task force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	No Action Needed. See item 2 above.
4	The task force recommends that the NRC strengthen SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events.	Not Applicable. See item 2 above.
5	The task force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees
6	The task force recommends, as part of the longer term review, that the NRC identify insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-ichi accident.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.
7	The task force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	Not Applicable.
8	The task force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs.	No Action Needed. Under severe external event conditions, dose effects to members of the public are not expected to exceed EPA protective action guidelines.

Recommendations		Review Result
9	The task force recommends that the NRC require that facility EPs address prolonged SBO and multiunit events.	No Action Needed. See item 8 above.
10	The task force recommends, as part of the longer term review, that the NRC should pursue additional EP topics related to multiunit events and prolonged SBO.	No Action Needed. See item 8 above.
11	The task force recommends, as part of the longer term review, that the NRC should pursue EP topics related to decision-making, radiation monitoring, and public education.	No Action Needed. See item 8 above.
12	The task force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the reactor oversight process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.

III. Conclusion and Recommendations

NRC staff concludes that the uranium recovery and mill tailing facilities licensed by the NRC or Agreement States are appropriate to the scope and potential hazard. The current assessment reviewed various external events to determine if a failure of safety system(s) or barrier(s) at uranium recovery facilities could cause significant release of radioactive materials that would harm workers or the public or cause significant damage to the environment. Potential hazards from external events are believed to be insignificant or low for the above facilities. Therefore, NRC staff concludes that no further study or regulatory action is necessary for uranium recovery facilities.

7. Decommissioning Reactors and Complex Materials Facilities

I. Current Regulatory Framework

The NRC regulates the decommissioning of power reactors, research and test reactors, materials and fuel cycle facilities, and uranium recovery facilities. The decommissioning program ensures that NRC-licensed sites are decommissioned in a safe, timely, and effective manner so that they can be returned to beneficial uses. Each year, the NRC terminates about 150 materials licenses. Most of these license terminations are routine, and the sites require little, if any, remediation to meet the NRC's unrestricted release criteria.

In general, most decommissioning research and test reactors, materials and fuel cycle facilities, and uranium recovery facilities, or sites, present low hazard or risk because radioactive sources, including spent fuel, are typically removed after operations cease when entering into the decommissioning mode. Decommissioning involves subsequent decontamination and remedial actions to reduce residual radioactivity to a level corresponding to the dose criteria of 10 CFR Part 20, Subpart E.

The potential radiological hazard at decommissioning facilities is normally low, as it is typically associated with residual radioactivity on equipment, surfaces, and surface or subsurface soil. Low hazard may also be associated with radioactive waste materials resulting from cleanup, decontamination, and dismantlement that may be transition activities before permanent disposal or disposition. Decontamination and treatment processes during decommissioning present minor hazards to workers.

Radionuclides remaining after cessation of operation are typically contained on surfaces of equipment or buildings, or in the soil. The potential for a release of the radioactive materials requires severe environmental or climate conditions such as flooding, tornado, or earthquake.

For power reactor decommissioning sites, the principal radiological risks are associated with the storage of spent fuel onsite. Most decommissioning sites have spent fuel stored only in independent spent fuel storage installations, which are addressed in Section 1 of this enclosure. Five power reactors recently permanently shut down and defueled and entered into a decommissioning status (Kewanee Power Station; San Onofre Generating Station Units 1 and 2; Crystal River Unit 3 Nuclear Generating Station; and Vermont Yankee Nuclear Power Station). These recently permanently shut down power reactors still have spent fuel stored in the spent fuel pool (SFP), as well as in independent spent fuel storage installations (except for Crystal River, which does not yet have an independent spent fuel storage installation). In addition, one other older permanently shutdown power reactor (Millstone Unit 1) still has fuel stored in the SFP. Note that several decommissioning power reactors no longer have any fuel stored onsite and the risks are the same as for the material decommissioning sites discussed below. The applicability of Fukushima Dai-ichi lessons learned to the decommissioning power reactor with fuel still stored in the SFP are addressed separately from the assessment of other decommissioning material sites.

Licensing

The NRC, its licensees, and the Agreement States share responsibility to protect public health and safety and the environment during decommissioning of reactors and material facilities. The decommissioning sites cover a variety of licensing activities including power reactors, research and test reactors, complex materials, uranium recovery, and fuel cycle facilities. There are about 49 sites in Agreement States undergoing decommissioning (36 materials and 13 uranium recovery facilities). NRC decommissioning sites include: (1) 18 power reactors, (2) 8 research and test reactors, (3) 16 complex non-reactor facilities, (4) 39 uranium recovery facilities, and (5) 2 fuel cycle facilities.

Regulations and Guidance

The following regulations specifically apply to facilities undergoing decommissioning:

- 10 CFR Part 20, Subpart E, "Standards for protection against radiation, radiological criteria for license termination"
- 10 CFR 30.36, byproduct materials: "Expiration and termination of licenses and decommissioning of sites and separate buildings, and outdoor areas"
- 10 CFR 40.42, source material: "Expiration and termination of licenses and decommissioning of sites and separate buildings, and outdoor areas"
- 10 CFR 40, Appendix A, "Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the Extraction or Concentration of Source Material from Ores Processed Primarily for Their Source Material Content"
- 10 CFR 50.82, production and utilization facilities: "Termination of license"
- 10 CFR 50.83, "Release of part of a power reactor facility or site for unrestricted use"
- 10 CFR 70.38, special nuclear material: "Expiration and termination of licenses and decommissioning of sites and separate buildings, and outdoor areas"
- 10 CFR 72.54, "Expiration and termination of licenses and decommissioning of sites and separate buildings, and outdoor areas"
- 10 CFR 72.218, "Termination of licenses"

The primary decommissioning guidance documents are NUREG-1757, "Consolidated Decommissioning Guidance, Volumes I-IV;" NUREG-1700, Rev. 1, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans;" NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM);" and NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors." These NUREGs describe: (1) methods acceptable to NRC staff in carrying out specific parts of the Commission's regulations; (2) techniques and criteria used by NRC staff in evaluating decommissioning actions; and (3) guidance to licensees responsible for decommissioning NRC-licensed sites.

The decommissioning process for materials licensees takes a risk-informed, performance-based approach and ensures compliance with the radiological criteria for license termination in 10 CFR Part 20, Subpart E. The approaches to license termination described in the regulatory guidance provide the information (subject matter and level of detail) needed to terminate a license by considering the specific circumstances of the range of radioactive materials users licensed by the NRC. Volume 1 is applicable to the decommissioning of materials facilities licensed under 10 CFR Parts 30, 40, 70, and 72 and to the ancillary surface facilities that support radioactive waste disposal activities licensed under 10 CFR Parts 60, 61, and 63. For complex material sites, a decommissioning plan is required to be developed within one year of site shutdown.

When a power reactor permanently shuts down, a post-shutdown decommissioning activities report (PSDAR) is provided by the licensee, as required by 10 CFR 50.82. Guidance on the information and content contained in the PSDAR is specified in Regulatory Guide 1.185, "Standard Format and Content for Post-Shutdown Decommissioning Activities Report." The purpose of the PSDAR is to provide the NRC and the public with a general overview of the licensee's proposed decommissioning activities and to inform NRC staff of the licensee's expected activities and schedule so that NRC staff can plan for inspections and make decisions about its oversight activities. The PSDAR is also a mechanism that informs the public of the proposed decommissioning activities before the conduct of those activities. Additional guidance on the power reactor decommissioning process is provided in NRC Regulatory Guide 1.184, "Decommissioning of Nuclear Power Reactors."

Most of the regulations applicable to operating power reactor licensees in 10 CFR Part 50 continue to apply to power reactors that have permanently shut down. Because of the reduced risk posed by decommissioning power reactor sites when compared to operating reactors, many licensing actions – primarily involving exemptions and amendments – have been submitted by the licensees and processed by NRC staff in order to establish a long-term licensing framework that is representative of the reactor's permanently shut down and defueled status. These license actions involve emergency planning, security, onsite and offsite insurance, operator staffing and training, and technical specifications. Decommissioning power reactor licensees maintain the licensing basis of the facility and reflect the status of the facility as decommissioning progresses in the updated final safety analysis report (UFSAR).

In addition, a license termination plan (LTP), must be submitted two years or more before license termination. The level of detail submitted in the LTP will vary depending on when the licensee submits the LTP. The information submitted in the LTP should reflect the current status of the decommissioning at the facility. The regulations in 10 CFR 50.82(a)(9)(ii) require that the LTP include the following information: site characterization, identification of the remaining dismantlement activities, plans for site remediation, detailed plans for the final radiation survey, description of the end use of the site, an updated site-specific estimate of remaining decommissioning costs, and a supplement to the environmental report describing any new information or significant environmental change associated with the licensee's proposed termination activities. The LTP must be submitted as a supplement to the licensee's final safety analysis report or as an equivalent document.

Inspection, Licensing, and Oversight

The goals of the inspection and licensing program at sites undergoing decommissioning are to:

- Ensure, through the review of the decommissioning submittals and surveys, that the decommissioning can be conducted safely and in accordance with NRC's regulations and requirements.

- Ensure, through direct observation and verification, that decommissioning is being conducted in accordance with the decommissioning activities as described in the respective decommissioning submittal and NRC's regulations.

- Ensure that corrective actions are taken by facilities undergoing decommissioning if they are not conforming to the decommissioning activities as described in the respective decommissioning submittal or NRC's regulations.

- Ensure that the site, at the completion of decommissioning, complies with NRC's criteria for license termination.

The NRC does not independently monitor radioactive releases on a continuous basis, but performs an independent measurement of the releases to verify the licensee's samples. The NRC also reviews licensee's documents and procedures for monitoring releases and reviews the results periodically. The inspection program, which contains the objectives and procedures for each type of inspection, is described in Inspection Manual Chapter (IMC) 2602, "Decommissioning Oversight and Inspection Program for Fuel Cycle Facilities and Material Licensees;" IMC 2545, "Research and Test Reactor Inspection Program;" and IMC 2561, "Decommissioning Power Reactors Inspection Program."

Emergency Preparedness

Emergency preparedness considerations for dismantlement and decontamination of facilities take into account the reduced risk of accidents, particularly for power reactors and fuel cycle facilities. As indicated above, risk or hazard would be greatly reduced for facilities undergoing decommissioning shortly after cessation of operations and upon entering into the decommissioning mode (e.g., after removal of spent fuel and/or radioactive sources).

For facilities transitioning into defueling, dismantling, decontamination, and decommissioning; emergency preparedness would overlap with decommissioning planning to assess and evaluate other safety issues and hazards. Under such conditions, a site-specific emergency preparedness plan would be developed. In this context, EPA's protective action guidelines would be used in the early and intermediate phases of an emergency.

For power reactors undergoing decommissioning, exemptions from certain emergency preparedness requirements for normal operation (e.g., 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50) may be granted. The NRC has recently approved emergency preparedness exemptions based on when the decay heat level of the spent fuel in the SFP has lowered to the point that the risk of a zirconium fire is negligible. Licensees are still required to maintain an onsite emergency plan addressing the classification of an emergency, and notification and coordination with designated offsite government officials following an event declaration. If needed, offsite authorities may implement protective actions using a comprehensive emergency management (all-hazard) approach to protect public health and safety. These licensees are no longer required to maintain formal offsite radiological emergency preparedness, including the 10-mile emergency planning zone.

As part of the process, the licensee submits an amendment to their emergency plan that conforms to the exemptions granted by the NRC. The licensee will continue to notify the NRC and the State following the declaration of an emergency classification, and maintain communications and interface responsibilities with Federal, State, and local organizations that may provide assistance in the event of an emergency declaration. In addition, provisions for fire, law enforcement, ambulance and medical services will continue to be provided via letters of agreement with local entities.

External Events

External events, human-induced or natural events, are typically evaluated in the preliminary phases of the site evaluation process.

The decommissioning plan (DP) typically addresses meteorology, climate, and surface water, as well as the hydrological system at the facility. The decommissioning process and plan also includes aspects of survey and monitoring, storage and control of licensee's material as well as decommissioning material not in storage (see NUREG-1757, Appendix H). Further, the DP includes auditing and surveillance. Notification and inspection, including criticality safety, are significant aspects of the decommissioning reviews. External events such as flooding, earthquakes, tornadoes, volcanic eruptions, or severe climate conditions are usually addressed in the design of the facility[24] and in the decommissioning plan (or UFSAR for decommissioning power reactors). In the unlikely event that such an external event were to occur, the status of the site is reported and a site-specific assessment of significant safety impacts is conducted to identify immediate actions that need to be taken to mitigate significant radiological releases.

Under 10 CFR 20.1406, licensees must describe how the facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, to facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. In addition, licensees should, to the extent practical, conduct operations to minimize the introduction of residual radioactivity into the site, including the subsurface, in accordance with the existing radiation protection requirements in Subpart B and the radiological criteria for license termination in 10 CFR Part 20 Subpart E.

For materials decommissioning sites the risk from external events is low, because of the low activity of the sources and the low potential of movement by external events for such sources. If such movement of sources at decommissioning materials facilities occurred, there would be dilution of such sources (e.g., from surface water or ground water, or mixing with clean soil) and potential dose effects would be less than public dose limits under 10 CFR Part 20. Under extreme external event conditions, NRC staff determined that the potential dose effects to members of the public would not exceed EPA protective action guidelines.

[24] For power reactors, 10 CFR Part 50.2 provides a definition for "design bases." For these facilities, "design basis events" consist of those events and accidents that the NRC has – as a matter of historical practice – required licensees to consider when identifying safety-related SSCs needed to provide key safety functions (see 10 CFR 50.49). For facilities other than power reactors regulated by the NRC, "design basis" may not be defined. Regardless, for the purposes of this assessment the staff focused its review on extreme external events that go beyond those that were considered in the licensing of the facility.

West Valley Demonstration Project

NRC staff received several stakeholder comments on a draft of its assessment. The draft assessment was issued to support a Category 3 public meeting on the topic. One of the stakeholder comments provided in response to the Category 3 public meeting (which was submitted jointly by the Alliance for a Green Economy, Citizens' Environmental Coalition, and the Nuclear Information and Resource Service), expressed concern regarding the assessment's applicability to the West Valley Demonstration Project (WVDP) in Western New York.

The WVDP is a unique case and different from other NRC-licensed complex sites. The WVDP Act of 1980 (WVDPA) gives the NRC an advisory and a monitoring role with respect to the WVDP activities. Under the WVDPA, the Department of Energy (DOE) has responsibility to decontaminate and decommission: (a) the tanks and other facilities of the Western New York Service Center in West Valley in which high-level waste was solidified or stored, (b) the facilities used in the solidification of waste, and (c) any material and hardware in connection with the project. As a result of the Act, the WVDP NRC license was put into abeyance. When the DOE's work under the Act is completed, the NRC license will be reactivated and the NRC licensee, New York State Energy Research and Development Authority (NYSERDA), will be responsible for the license.

Nevertheless, until the WVDP NRC license is reactivated, the NRC responsibilities under the WVDPA are limited. NRC responsibilities under the WVDPA are to provide decommissioning criteria, informal review and comment, and monitoring. The objective of the monitoring activities is to help provide the bases for the Office of Nuclear Material Safety and Safeguards to advise the DOE of whether activities conducted at the WVDP facilities and their operation are adequate to protect the public health and safety and the environment from radiological hazards. However, protection of public health and safety and the environment is the sole responsibility of the DOE and its contractors.

The WVDP decommissioning will be conducted in two phases. NRC staff provided a technical evaluation report to the DOE on February 25, 2010, documenting the staff's review and comment on Phase I of the WVDP decommissioning plan (ADAMS Accession No. ML100360030). The key objective of the staff's review of this plan was a determination on whether the proposed action described in the Phase 1 decommissioning plan satisfies the decommissioning criteria for unrestricted use specified in 10 CFR 20.1402. Based on its review, NRC staff determined that the plan provides reasonable assurance that the proposed action will meet the decommissioning criteria.

Natural phenomena and external events were addressed in DOE's Final Environmental Impact Statement (EIS) which was issued in January 2010 (ADAMS Accession No. ML100750122). In addition, an earlier Safety Analysis Report (SAR) addressed external events in more detail (see for example SAR document ID Number WVNS-SAR-023 (ADAMS Accession No. ML032940574). The NRC may informally review documents associated with WVDP activities as a result of its monitoring activities or it may do so as a result of commitments that DOE made in its WVDP Phase I Decommissioning Plan, or as a result of learning about development of DOE documents in NRC's informal consultancy role. Phase II of the WVDP Decommissioning Plan is not yet developed. DOE and NYSERDA plan to develop the supplemental EIS by 2020. A Decommissioning Plan for Phase II will also be developed.

The primary responsibility for assessing Fukushima lessons learned with respect to the WVDP lies with DOE. The NRC recently became aware of a DOE report published in December 2014 (http://energy.gov/sites/prod/files/2014/12/f19/2014_WVDP_Emergency_Mgmt_Tech._Basis_and_Emerg._Preparedness-December%20_2014.pdf) that assessed the DOE-WVDP's progress on implementing the Fukushima lessons learned into its Emergency Management Program. Through the DOE Office of Enterprise Assessments (EA), a review was performed in part to determine WVDP's progress in implementing enhancements recommended in DOE policy that stemmed from the DOE's evaluation of Fukushima Lessons Learned for its facilities (OE-1, "Improving DOE Capabilities for Mitigating Beyond Design Basis Events (BDBEs)."

The DOE review found, among other things, that the method DOE used to quantify the material at risk in one of its storage areas was not adequate, and that the WVDP plans and procedures may not support appropriate and conservative event categorization and classification, protective actions, and offsite notification. Due to the nature of some of the material that must be stored, handled, and transported during decommissioning, the area of emergency response may need to be enhanced. NRC staff will continue to monitor this situation.

II. Evaluations and Assessments Prior to Fukushima

Before Fukushima, NRC staff evaluated the potential effects of certain external events on nuclear power reactor facilities that had permanently ceased operation and defueled. The concern was for facilities where irradiated fuel remains in the SFP. In this context, two analyses and studies were conducted for SFPs subjected to postulated beyond-design-basis external events. NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," contains an evaluation of the potential accident risk in a spent fuel pool at decommissioning plants in the United States. This study was prepared to supply a conservative technical basis for decommissioning rulemaking for permanently shut-down nuclear power plants. It describes:

- the modeling approach of a typical decommissioning plant with design assumptions and industry commitments,
- the thermal-hydraulic analyses performed to evaluate the behavior of spent fuel stored in the SFP at decommissioning plants,
- the risk assessment of SFP accidents,
- the consequence calculations, and
- the sensitivity study and implications for decommissioning regulatory requirements.

NUREG-1738 concluded that: "Given the robust structural design of SFP, it is expected that a seismic event with peak spectral acceleration several times larger than the SSE [safe shutdown earthquake] would be required to produce catastrophic failure of the structure. The estimated frequency of events of this magnitude differs greatly among experts and is driven by modeling uncertainties." In summary, the results of the study indicate that the risk at SFPs is low and well within the Quantitative Health Objectives. The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious. The study includes use of a pool performance guideline as an indicator of low risk at decommissioning facilities.

In 2005,[25] NRC staff studied SFPs at Zion 1 and 2, Millstone Power Station 1, La Crosse BWR, Humboldt Bay 3, Indian Point 1, and GE Morris. The study concluded that

> it is not necessary to develop additional mitigation strategies for drain-down events for the five decommissioning power reactor SFPs. The heat-up is slow, because of the long decay time of the most recently discharged fuel in these SFPs. The minimum decay time of the fuel in these SFPs is 8.5 years. The decay power for 8.5 year old fuel is a factor of six lower than 1 year old fuel. Because of the slow heat-up time associated with such long decay times, current mitigation strategies augmented by ad hoc mitigation strategies are adequate. Also there appears to be no way to gravity drain the GE Morris SFP, even by introducing a hole in its side or floor.

More recently, NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," dated September 2014 (ADAMS Accession No. ML14255A365), was issued. The study provides consequence estimates of a hypothetical spent fuel pool accident initiated by a low likelihood seismic event at a specific reference plant. The study compares high-density and low-density loading conditions and assesses the benefits of post-9/11 mitigation measures. Past risk studies have shown that storage of spent fuel in a high-density configuration is safe and risk of a large release due to an accident is very low. This study's results are consistent with earlier research conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking.

III. Post Fukushima Event Evaluations/Assessments

<u>Assessment for Facilities Undergoing Decommissioning Other than Power Reactors</u>

The current assessment reviewed various external events to determine if a failure of safety barrier(s) at decommissioning facilities could cause significant release of radioactive materials to harm workers or the public or cause any significant damage to the environment. The nature of events assessed and initial evaluation are summarized in the table below. The potential dose impact to members of the public under extreme external events do not exceed EPA protective action guideline levels under emergency situation.

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[25] The 2005 study assessed the risk at decommissioning these power plants and GE Morris due to catastrophic failure of the spent fuel pool. While the study contains sensitive information the conclusion is not.

Effect of External Events on Decommissioning Facilities

<table>
<tr><th>External Event</th><th>Outcome</th><th>Assessment</th></tr>
<tr><td>Flood</td><td>Challenge to structures at decommissioning facilities which may contain residual radioactive materials higher than release limits on surfaces.

Challenge to areas with residual radioactivity above release limits in soil.

Challenge to areas storing decommissioning waste in a transition for onsite disposal or ultimate off site disposition.</td><td rowspan="9">During the decommissioning process licensees are required to submit either a “Decommissioning Plan” or a “License Termination Plan” describing all safety aspects to protect workers, the public, and the environment from potential release of radioactivity. In addition, as indicated above, licensees are required to assess site location, climate conditions, and potential safety issues related to certain events such as extreme rain or flooding conditions, tornadoes, and high wind events. During decommissioning, radioactive materials are either on building surfaces, in soil, or could be stored in piles or in containers for ultimate packaging and disposition.

NRC staff does not expect radioactive materials to be released from decommissioning facilities that exceed Category 1 or Category 2 sources (e.g., sources that must be reported to and tracked in accordance with 10 CFR 20.2207). For decommissioning facilities other than decommissioning power reactors emergency plans that address natural phenomena, including an earthquake, a tornado, flooding, or other phenomena, generally are not required, though external events are addressed in safety guides.

Under extreme event conditions, NRC staff determined that the potential dose effects to members of the public would not exceed EPA protective action guidelines.</td></tr>
<tr><td>Seismic</td><td>Challenge to structures in which radioactive materials are stored or present as residual radioactivity above release limits.</td></tr>
<tr><td>High Wind and Missiles</td><td>Challenge to structures in which radioactive material is present above release limits or stored in a transition for disposal.</td></tr>
<tr><td>Lightning</td><td>Challenge to structures that may catch fire during lightning strikes which may contain stored radioactive materials in transition before disposition.</td></tr>
<tr><td>Snow and Ice Loads</td><td>Challenge to structures in which radioactive material is stored or being treated for disposition.</td></tr>
<tr><td>Drought</td><td>None</td></tr>
<tr><td>Temperature Extremes</td><td>None except for pyrophoric materials stored before disposition.</td></tr>
<tr><td>External Fire</td><td>Challenge to structures in radioactive materials stored in a transition for disposition.</td></tr>
<tr><td>Loss of Power</td><td>None</td></tr>
</table>

NRC staff considered the 12 NTTF recommendations for applicability to decommissioning facilities other than power reactors to determine whether future action is warranted. The following table summarizes the result of the review.

Near-Term Task Force Recommendations and Future Actions

Recommendations		**Review Result**
1	Establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.
2	The task force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection of SSCs.	No Action Needed. For research and test reactors, complex non-reactor facilities, uranium recovery facilities, and two fuel cycle facilities undergoing decommissioning, the risk from external events is low because of the low activity of the sources and the low potential of movement by external events for such sources. Under extreme event conditions, NRC staff determined that the potential dose effects to members of the public would not exceed EPA protective action guidelines.
3	The task force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	No Action Needed. See item 2 above.
4	The task force recommends that the NRC strengthen SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events.	No Action Needed.
5	The task force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	No Action Needed. For research and test reactors, complex non-reactor facilities, uranium recovery facilities, and two fuel cycle facilities undergoing decommissioning scenarios that create hydrogen in quantities of concern are not credible.
6	The task force recommends, as part of the longer term review, that the NRC identifies insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the	No Action Needed. For research and test reactors, complex non-reactor facilities, uranium recovery facilities, and two fuel cycle facilities undergoing decommissioning scenarios that create hydrogen in quantities of concern are not credible.

	Recommendations	Review Result
	Fukushima Dai-ichi accident.	
7	The task force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	No Action Needed. Research and test reactor facilities undergoing decommissioning have had their spent fuel removed from the facility. For complex non-reactor facilities, uranium recovery facilities, and two fuel cycle facilities undergoing decommissioning this lessons learned item is not applicable.
8	The task force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs.	No Action Needed. See item 2 above.
9	The task force recommends that the NRC require that facility EPs address prolonged SBO and multiunit events.	No Action Needed. See item 2 above.
10	The task force recommends, as part of the longer term review, that the NRC should pursue additional EP topics related to multiunit events and prolonged SBO.	No Action Needed. See item 2 above.
11	The task force recommends, as part of the longer term review, that the NRC should pursue EP topics related to decision-making, radiation monitoring, and public education.	No Action Needed. See item 2 above.
12	The task force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the reactor oversight process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.

Assessment for Power Reactors Undergoing Decommissioning

As discussed above, there are 18 power reactors undergoing decommissioning. Five of these reactors have recently shutdown permanently and removed fuel from their reactor vessels: Crystal River Unit 3, Kewaunee, San Onofre 2 and 3, and Vermont Yankee. These reactors still have spent fuel stored in SFPs, as well as in independent spent fuel storage installations (ISFSIs) (except for Crystal River, which does not yet have an ISFSI). Of the remaining 13 power reactors undergoing decommissioning, only one of the reactors (Millstone Unit 1) stores spent fuel in a SFP. The remaining power reactors undergoing decommissioning either have their spent fuel stored in an ISFSI, in an operating unit's SFP, or the fuel has been shipped offsite.

Power reactors undergoing decommissioning have two risks associated with them: (1) control of radioactive material other than spent fuel (similar to the decommissioning of other material sites) and (2) proper storage, protection, and handling of spent fuel (for those facilities that still have fuel onsite in either an independent spent fuel storage installation or SFP). Regarding the risks associated with the control of radioactive material other than spent fuel, the assessment found above for decommissioning material facilities is considered applicable to power reactors undergoing decommissioning. This assessment concludes that additional regulatory action is not necessary based on Fukushima lessons learned for these facilities.

For power reactors undergoing decommissioning that store their spent fuel in ISFSIs, the assessment found in Section 1 of this enclosure is applicable. This assessment concludes that additional regulatory action is not necessary based on Fukushima lessons learned for these facilities. If the spent fuel is stored in an operating power reactor SFP, post-Fukushima actions taken for operating power reactors are applicable and safety is assured through those actions.

<u>Assessment for Decommissioned Power Reactors with Irradiated Fuel Stored in Spent Fuel Pools</u>

As described above the risk associated with these facilities is associated with loss of cooling to the spent fuel leading to cladding failure and offsite release. For the five power reactors that have recently permanently shutdown and defueled their reactors, the orders associated with Fukushima lessons learned were rescinded. Specifically, the following orders were rescinded for these five power reactors:

- Order EA-12-051, which requires reliable SFP instrumentation be installed and maintained. This order was issued to ensure that licensees were able to monitor SFP level to appropriately prioritize mitigation and recovery actions during an event.

- Order EA-12-049, directing licensees to develop and implement strategies to maintain or restore core cooling, containment cooling, and spent fuel cooling capabilities following a beyond-design-basis external event. Compliance with this order includes an initial phase, a transition phase, and a final phase.

In addition, Order EA-13-109 was rescinded for the lone boiling water reactor (Vermont Yankee) that is included in the group of five power reactors that have permanently shutdown recently. Order EA-13-109 requires that all licensees that operate boiling water reactors with Mark I and Mark II containment designs, including the Vermont Yankee Nuclear Power Station, implement requirements for reliable hardened containment vents capable of operation under severe accident conditions at their facilities.

The basis for rescinding Order EA-12-051 is that because the licensees certified the permanent removal of fuel from the reactor vessel, the SFP becomes the primary safety concern 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 response personnel would be on the SFP. Thus, the basis for this order no longer applies.

The basis for rescinding Order EA-12-049 for these units includes:

- Guidance and strategies to maintain or restore core cooling and containment are no longer necessary because the nuclear fuel has been permanently removed from the reactor vessels and containments.

- The decay heat levels in the SFPs at the time the unit would have been required to be in compliance with the order are low. At these low decay heat levels, reliance on the SFP coolant inventory for passive cooling will provide an equivalent level of protection to that required by the initial phase of Order EA-12-049.

- At the time compliance with the order would have been required, the low decay heat levels will provide a long time to boil off the SFP inventory, and consequently a long time to reach a point at which makeup would be necessary for radiation shielding purposes. This eliminates the need for a transition phase of Order EA-12-049, where makeup water would be added using portable onsite equipment.

- The low decay heat and long boil-off period at the time compliance with the order would have been required provides sufficient time for the licensee to obtain off-site resources on an ad hoc basis to sustain the SFP cooling function indefinitely, eliminating the need for the final phase required by Order EA-12-049.

The basis for rescinding Order EA-13-109 for Vermont Yankee is found in an April 15, 2014, letter (ADAMS Accession No. ML14055A323). The basis states that following submittal of the 10 CFR 50.82(a)(1) certifications for permanent cessation of operations and the permanent removal of fuel from the reactor vessel, the primary containment will no longer provide a safety function and the underlying purpose of Order EA-13-109 will no longer be served. On January 12, 2015, the Vermont Yankee licensee provided the certifications for permanent cessation of operations and the permanent removal of fuel from the reactor vessel (ADAMS Accession No. ML15013A426).

In addition to the rescission of the orders, NRC staff has determined that the March 12, 2012, 10 CFR 50.54(f) request for information for flooding and seismic reevaluations and for emergency planning applicable to operating reactors is generally not applicable to the five power reactors that have recently permanently shutdown and defueled their reactors, for the following reasons:[26]

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[26] The reasons given hear are generally applicable to the five reactors that have recently permanently ceased operations. The basis for the March 12, 2012, request for information responses not being necessary for San Onofre Units 2 and 3 and Crystal River Unit 3 are found in letters dated January 22, 2014 (ADAMS Accession Nos. ML13329A826, and ML13325A847, respectively). The staff documented that Kewaunee is no longer obligated to respond to the March 12, 2012, request for information in letters dated October 9, 2013, and January 22, 2014 (ADAMS Accession Nos. ML13261A128, and ML13322B255, respectively). The staff is currently considering a March 12, 2015, Entergy letter providing a position that the March 12, 2012, request for information is no longer applicable to Vermont Yankee.

- No further responses or actions associated with the 10 CFR 50.54(f) letter are necessary, because the licensee is no longer authorized to load fuel into the reactor vessel and potential fuel-related accident scenarios are limited to the SFP. Unlike the reactor, the safety of fuel located in the SFP is assured for an extended period through maintenance of pool structural integrity, which preserves coolant inventory and maintains margin to prevent criticality. Small changes in the flooding hazard elevation would not threaten the structural integrity of a flooded pool. Further, previous evaluations of SFP structures have determined that seismic margins are very large. As seismic and flooding studies continue for the remainder of the operating fleet, new information concerning the adequacy of design bases of SFPs will be evaluated for applicability to decommissioned sites using existing NRC processes.

- Based on the discussion above, the safety of the fuel stored in SFPs would not be substantially affected by potential changes in the flooding or seismic hazard levels. Furthermore, for beyond design basis external events challenging the safety of the spent fuel, 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, SFP 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 10 CFR 50.54(f) letter addresses.

NRC staff also assessed the Millstone Unit 1 SFP. The logic for rescission of the Orders EA-12-051 and EA-12-049, and the 10 CFR 50.54(f) request for information, to the five recently permanently shutdown and defueled power reactors is also the basis for why these orders and request for information did not apply to Millstone Unit 1. Millstone Unit 1 permanently shutdown on July 21, 1998, so the decay heat load associated with fuel in its spent fuel pool is a fraction of that found in the reactors that recently entered the decommissioning mode. In addition, although a boiling water reactor, Order EA-13-109 associated with Mark I and Mark II containments is not applicable to Millstone 1, because the fuel is in the SFP and the containment no longer provides a safety function.

Additional Capabilities at Recently Permanently Shutdown and Defueled Power Reactors

As previously noted, NUREG-1738 and NUREG-2161 suggest that the radiological consequences of a postulated beyond-design-basis accident that results in a zirconium fire of the fuel stored in the spent fuel pool can be extensive; however, the likelihood of such an accident is very low, even considering the most severe natural phenomena. In addition, for SFPs with sufficient decay time, there is little potential to cause offsite early fatalities regardless of the type of offsite response. Furthermore, even though the possibility of a beyond-design-basis SFP accident that results in a significant loss of SFP cooling water inventory is very unlikely, the recently decommissioned reactors have multiple strategies for providing makeup to the SFP which include, for example: using existing plant systems for inventory makeup, supplying water through hoses to a spool piece connection to the existing SFP piping, or using a diesel-driven portable pump to take suction from other sources of water to provide makeup or spray to the SFP. NRC staff has assurance that these strategies will continue to be maintained at the five recently permanently shutdown power plants based on a license condition for mitigation strategies that was established in response to the terrorist attacks of September 11, 2001, and was maintained even after these facilities permanently ceased operation.

For this and other reasons, the five recently permanently shutdown power plants have proposed exemptions that would result in elimination of the requirements for formal offsite radiological emergency plans at these sites (but would require the maintenance of certain onsite capabilities to communicate and coordinate with offsite response authorities). Before granting the requested emergency planning exemptions, NRC staff has sought Commission approval. In general, exemptions to certain emergency planning requirements have been granted when the site-specific analyses show that at least 10 hours is available for mitigative actions from a postulated beyond-design-basis SFP drain-down event where cooling of the spent fuel is not effective. The assumptions include beyond-design-basis accidents that result in either a partial drain-down or the complete drainage of the SFP with rearrangement of spent fuel rack geometry and/or the addition of rubble to the SFP that would effectively impede any decay heat removal through all possible modes of cooling. The analysis postulates that decay heat transfer from the spent fuel via conduction, convection, or radiation would be impeded. This analysis is often referred to as an adiabatic heatup. The analysis demonstrates that, assuming a loss of all modes of cooling for the spent fuel stored in the SFP, there would still be a minimum of 10 hours for the hottest fuel assembly to reach 900°C, which is the critical temperature threshold for self-sustained oxidation of zirconium cladding in air. This will ensure that sufficient time exists to initiate appropriate mitigating actions.

The Commission has determined that the risk of offsite radiological release from these facilities is sufficiently low to exempt the licensee from certain requirements related to the need to maintain offsite emergency planning. The Commission has approved exemptions from certain emergency planning requirements through Staff Requirements Memorandum for San Onofre Units 2 and 3, Vermont Yankee, Crystal River Unit 3, and Kewaunee (ADAMS Accession Nos. ML15061A521, ML15061A516, ML14364A111, and ML14219A366, respectively).

Longer-Term Actions

On December 30, 2014, the Commission directed NRC staff to proceed with rulemaking on decommissioning (SRM-SECY-14-0118 at ADAMS Accession No. ML14364A111). This rulemaking will address issues discussed in SECY-00-0145, such as: (1) the graded approach to emergency preparedness, (2) lessons learned from the plants that have already (or are currently) going through the decommissioning process, (3) the advisability of requiring a licensee's PSDAR to be approved by the NRC, (4) the appropriateness of maintaining the three existing options for decommissioning and the timeframe associated with those options, (5) the appropriate role of State and local governments and non-governmental stakeholders in the decommissioning process, and (6) any other issues deemed relevant by NRC staff. Therefore, based on Commission direction, issues pertaining to emergency preparedness safety, defense-in-depth, and potential risk from external events would be anticipated to be addressed in more detail in the context of the anticipated rulemaking for decommissioning.

The table below summarizes the external events reviewed by NRC staff, the impact on the facility and a summary of the basis for no additional action being needed for this class of decommissioned power reactors.

Effect of External Events on Decommissioned Power Reactors

External Event	Outcome	Assessment
Flood	Challenge to spent fuel pool cooling and to spent fuel pool integrity.	• No assessment of the reactor or containment systems are necessary because the licensee is no longer authorized to load fuel into the vessel and potential fuel-related accident scenarios are limited to the spent fuel pool. • The safety of fuel located in the spent fuel pool is assured for an extended period through maintenance of pool structural integrity, which preserves coolant inventory and maintains margin to prevent criticality. • Exemptions from certain emergency planning requirements have been granted based on the low likelihood that spent fuel would reach the zirconium ignition temperature in less than 10 hours, which was determined using conservative assumptions. These assumptions include beyond-design-basis accidents that result in either a partial drain-down or the complete drainage of the SFP with rearrangement of spent fuel rack geometry and the addition of rubble to the SFP that would effectively impede any decay heat removal through all possible modes of cooling. • Recent and past studies have concluded that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking • Decay heat levels have decreased with time since these reactors permanently shut down such that
Seismic	Challenge to spent fuel pool cooling, spent fuel pool integrity, and spent fuel pool geometry.	
High Wind and Missiles	Challenge to spent fuel pool cooling, spent fuel pool integrity, and spent fuel pool geometry.	
Lightning	Challenge to spent fuel pool cooling.	
Snow and Ice Loads	Challenge to spent fuel pool cooling and to spent fuel pool integrity.	
Drought	Challenge to spent fuel pool cooling.	
Temperature Extremes	Challenge to spent fuel pool cooling.	
External Fire	Challenge to spent fuel pool cooling.	
Loss of Power	Challenge to spent fuel pool cooling.	

External Event	Outcome	Assessment
		the boil off of the SFP inventory would take days, and consequently, there is sufficient time to restore SFP cooling or implement other mitigating actions. • Recently decommissioned reactors have additional capabilities to provide makeup water and sprays to the SFPs that are required by a license condition.

NRC staff considered the 12 NTTF recommendations for applicability to decommissioning power reactors to determine whether future action is warranted. The following table summarizes the result of the review.

Near-Term Task Force Recommendations and Future Actions

Recommendations		Review Result
1	Establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment. Based on Commission direction, issues pertaining to emergency preparedness safety, defense-in-depth, and potential risk from external events are anticipated to be addressed in more detail in the context of the anticipated rulemaking for decommissioning.
2	The task force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection of SSCs.	No Action Needed. The 10 CFR 50.54(f) request for information is no longer applicable in light of the movement of the fuel to SFPs, the robust nature of the SFPs, and the low decay heat levels. Issues could be revisited based on lessons learned from the analysis of response from operating reactors.
3	The task force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	No Action Needed.
4	The task force recommends that the NRC strengthen SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events.	No Action Needed. Order-12-049 was rescinded for five reactors that have recently permanently ceased operation. Basis for order rescission is provided above, including fuel being moved to SFPs and low decay heat levels.

Recommendations		Review Result
5	The task force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	No Action Needed. Nuclear fuel has been permanently removed from the reactor vessels and containments.
6	The task force recommends, as part of the longer term review, that the NRC identifies insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-ichi accident.	No Action Needed. Spent fuel is stored in SFPs that are robust in nature. Low decay heat levels provide for time to restore cooling and prevent hydrogen generation. Issues could be revisited based on lessons learned from the analysis for this item for operating reactors.
7	The task force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	No Action Needed. Once fuel is in the SFP, the SFP becomes the primary safety concern 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 response personnel would be on the SFP condition. Thus, the basis for this recommendation and Order EA-12-051 no longer applies. Recently decommissioned reactors have additional capabilities to provide makeup water and sprays to the SFPs that are required by a license condition.
8	The task force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs.	No Action Needed. Based on SFPs being the focus of licensee's attention, the robust nature of designs, and low decay heat loads, 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.
9	The task force recommends that the NRC require that facility EP address prolonged SBO and multiunit events.	No Action Needed. See item 8 above.
10	The task force recommends, as part of the longer term review, that the NRC should pursue additional EP topics related to multiunit events and prolonged SBO.	No Action Needed. See item 8 above.

Recommendations		Review Result
11	The task force recommends, as part of the longer term review, that the NRC should pursue EP topics related to decision-making, radiation monitoring, and public education.	No Action Needed. See item 8 above.
12	The task force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the reactor oversight process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment. Based on Commission direction, issues pertaining to emergency preparedness safety, defense-in-depth, and potential risk from external events are anticipated to be addressed in more detail in the context of the anticipated rulemaking for decommissioning.

IV. Conclusion and Recommendations

For research and test reactors, complex non-reactor facilities, uranium recovery facilities, and the two fuel cycle facilities undergoing decommissioning, the current assessment reviewed various external events to determine if a failure of safety barrier(s) at these facilities could cause a significant release of radioactive materials that would harm workers or the public or cause significant damage to the environment. NRC staff concluded that these facilities undergoing decommissioning are appropriately licensed, given the radiological risk and potential hazards. Potential hazards for these facilities from external events are believed to be insignificant or low because the potential dose impact to the members of the public under extreme external events do not exceed EPA protective action guideline levels under an emergency situation. Therefore, no further study or regulatory action is recommended for such facilities.

The conclusion found above for decommissioning material facilities is considered applicable to power reactors undergoing decommissioning regarding the risks associated with the control of radioactive material other than spent fuel. The staff also assessed the risk of external events for the decommissioned power reactors that have fuel stored in their SFPs (including those that have recently permanently ceased operation). Previous studies and analyses (e.g., NUREG-1738 and NUREG-2161) have shown that the SFP structure is extremely robust and capable of withstanding the external events addressed by this paper. In addition, based on the decay heat levels of the five recently permanently shutdown reactors and the time available to take mitigating actions, there are no identified safety concerns that need further analysis. Further, as directed by the Commission in the staff requirements memorandum for SECY-14-0118, NRC staff is developing a comprehensive power reactor decommissioning rulemaking that will address the technical basis for an appropriate long-term regulatory framework for permanently shutdown reactors with fuel stored in the SFPs. Among other things, this rulemaking will likely examine the appropriate timing of certain regulatory transition points where current operating reactor requirements can be reduced based on the risks posed by external events in conjunction with thermal-hydraulic criteria that would be a function of the decay time of the spent fuel.

8. NRC-Licensed Non-Power Reactors

I. Introduction

NRC staff evaluated the 31 NRC-licensed research and test reactors (RTRs) to assess applicability of Fukushima Dai-ichi lessons learned. The RTRs were categorized into two categories based on the RTR's licensed thermal power. Twenty six RTRs licensed for less than 2 MW_t comprised Category 1, and the five RTRs greater than 2 MW_t comprised Category 2. The assessment concluded that all of Category 1 and two 2 MW_t research reactors from Category 2 were highly resilient to the loss of electrical power, active decay heat removal systems, and heat sink. The robust nature of these reactors is the result of their minimal decay heat generation – air cooling is sufficient to remove decay heat to prevent fuel failure. The three largest of the Category 2 reactors, two research and one test reactor, do not share the same level of resilience as the Category 1 reactors because of their reliance on water for adequate decay heat removal. NRC staff has concluded that additional assessment is needed to determine the resilience of the primary coolant system integrity to a beyond-design-basis seismic event for the research reactors and resilience of emergency power, active decay heat removal, and coolant make up systems to flooding scenarios and to a beyond-design-basis seismic event for the test reactor. However, the assessment of resilience for the three largest facilities is not urgent, because they do not have significant off-site consequences even under the most severe assumptions of damage. Such damage was assumed in the post-9/11 security assessments of multiple sabotage scenarios that considered the potential radiological consequences resulting from the severe damage to reactor fuel and facility containment or confinement systems.

II. Background

Non-power reactors (NPRs) are designed for many purposes, including commercial, research, testing, and education. These reactors are not used to produce electricity, process steam, or for desalination. All 31 operating NRC-licensed NPRs are used for research, testing, or education in nuclear engineering, physics, chemistry, biology, anthropology, medicine, materials sciences, and related fields and are all classified as RTRs. Among the RTRs that the NRC has licensed, the range of licensed thermal power varies from five watts to 20 megawatts-thermal (MW_t). There are also eight RTRs shut down and in various stages of decommissioning. This assessment considered only the 31 RTRs with current operating licenses. The eight RTR facilities in the decommissioning phase are included in the assessment described in Section 7.

The predominant radiological hazard associated with RTRs is radiological exposure from the mishandling of radioactive materials or experiments by operators or researchers. These materials or experiments may or may not contain nuclear fuel and can be located inside or outside the core in experimental facilities. The neutron and gamma ray beams, as well as byproduct materials produced by the activation of non-radioactive materials, may present radiological hazards. These materials typically do not present a hazard to the public due to the licensing limitations for experiments. The greatest hazard by far to the public results from the accumulation of fission products resulting from the fission of the uranium fuel. These radioactive materials are normally contained within the fuel cladding. In most cases, the reactor fuel is inside the core within pools of water to shield personnel from radiation and to remove heat produced from nuclear reactions. The reactors are housed within confinement or containment buildings. Therefore, an uncontrolled release of radioactive material could happen only in a case where both a loss of the fuel cladding integrity and the failure of the confinement controls or the containment barriers occur simultaneously.

The nature of the hazard and the mechanisms necessary for the accidental release of radioactive material from an RTR is similar to that of a power reactor. The difference between the current NRC-licensed RTRs and power reactors is the magnitude of their potential accident consequences. The difference in consequences results from the RTR's significantly different operating characteristics including maximum operating power level, temperatures, pressures, and the duration and frequency of operation. These operating characteristics result in significantly smaller fission product inventories for RTRs because of the significantly smaller quantity of fuel at the facility. To put these differences into perspective, consider the following: (1) the maximum licensed thermal power level of the smallest power reactor is about two orders of magnitude greater than that of the largest RTR; (2) the duration of operations of a power reactor is nearly continuous, while most RTRs are operated only periodically, typically a fraction of the normal work week; and (3) the onsite accumulation of significant quantities of spent fuel (20 to 40 years' worth) in fuel pools at power reactors as compared to little or no spent fuel at RTRs. These factors contribute to a significantly smaller nuclear material inventory, accident source term, and demand for active decay heat removal at RTRs.

III. Licensing

The NRC's authority to license and regulate NPRs is provided in Sections 103 and 104 of the Atomic Energy Act (Act). Section 103 of the Act pertains to the licensing of industrial or commercial reactors that can consist of both power and NPRs. Section 104 of the Act relates to the licensing of NPRs for the purpose of medical therapy and research and development. All RTRs currently licensed by the NRC are licensed under Section 104 of the Act. Unique to this authority are the provisions contained in Paragraph 104c of the Act that directs the "Commission to impose the minimum amount of such regulation and terms of license that will permit the Commission to fulfill its obligation under this Act to promote the common defense and security and to protect the health and safety of the public with the intent to permit the conduct of widespread and diverse research and development."

RTRs have been licensed under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," using the concept of defense-in-depth (DID). The concept of DID was applied at initial licensing to compensate for recognized uncertainties at the time (1950s and 1960s) related to nuclear reactor design, operation, and consequences associated with potential accidents. As such, a comprehensive DID approach forms the foundation for the design and licensing of all RTRs. Even with the accumulation of many reactor-years of operating experience and the development of more advanced analytical capabilities for the assessment of safe reactor operation and reactor accident consequences, the concept of DID remains as a relevant and effective means to address uncertainties.

Also, the implementation of 10 CFR Part 50, as it applies to RTRs, has been achieved using only deterministic methods and acceptance criteria. Inherent in these methods and criteria are the inclusion of highly conservative safety margins.

Similar to power reactors, a set of RTR licensing-basis events were established that are intended to ensure conservatism in design and protection from a wide spectrum of postulated accidents. Those accidents are highly stylized and do not consider multiple failures of safety systems. Qualitative approaches for ensuring reliable safety systems, such as the single failure criterion, were put in place. Testing plans and operational limits are established in technical specifications to ensure a high degree of confidence that safety systems would accomplish their designed safety functions if called upon during an accident.

It is common that the analysis of a set of postulated accidents for RTRs do not result in a radiological release. In order to assess the dose impact to the public, an incredible but hypothetical event that results in a radiological release is assumed to occur. This event must bound all the credible hazards resulting from the postulated accidents and is referred to in the siting and licensing of RTRs as the maximum hypothetical accident (MHA). The MHA assumes a failure of the fuel or a fueled experiment that results in radiological consequences (a release of radioactive material) that exceed those of credible accidents. The MHA is not expected to occur; therefore, only the potential consequences are analyzed and not the initiating event or scenario details. Guidance for the licensing of RTRs is provided in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," (Volume 1, "Format and Content," and Volume 2, "Standard Review Plan and Acceptance Criteria").

Licensing acceptance criteria allowing construction and permitting operation of a research reactor (not a test reactor) are based on meeting conservative assurance that the public and occupational dose resulting from the radiological release assumed in the MHA remain bounded by the normal (non-accident) occupational and public dose limits contained in 10 CFR Part 20, "Standards for protection against radiation." When the occupational or public dose resulting from the release from a research reactor exceeds the occupational and public dose limits of Part 20, the staff conducts additional review to ensure that the supporting MHA analysis is based on conservative assumptions.

The siting and licensing of a test reactor also requires the development of an MHA from which a postulated radiological release is calculated. For the test reactor only, the public and occupational dose resulting from the release must meet 10 CFR Part 100, "Reactor site criteria."

The effects of external events are also considered specifically in the RTR licensing process. NRC-licensed RTRs are required to demonstrate, in their design, reasonable assurance that external events would not prevent safe operation and shutdown of the reactor. They must also demonstrate that provisions are included to mitigate or prevent an uncontrolled release of radioactive material. The consequences of external events are considered and bounded by analyzed accidents, particularly the MHA. The traditional licensing approach for the consideration of external events for RTRs relies on documented historical averages and extremes, credible event frequencies, and predictive potential for the specific external events. At a minimum, each research reactor facility is required to meet the local building codes for the specific type of event (e.g., seismic, flooding) and test reactors are required to meet the requirements of 10 CFR Part 100.

IV. Inspection and Oversight

NRC staff inspects each facility periodically to ensure that licensees safely conduct regulated activities and maintain their facility in compliance with regulatory requirements. The NRC employs a graded inspection program for operating RTRs with two separate levels of inspection based on the maximum licensed power level of the facility. The scope of the inspection program for RTRs licensed to operate at power levels of 2 MW_t or greater, is more comprehensive and completed annually, whereas the inspection program for RTRs licensed to operate at power levels below 2 MW_t is completed every two years. Inspection programs for RTRs include: organizational structure, qualifications and responsibilities, operational activities, design and design control, review and audit functions, radiation and environmental protection, operator requalification, maintenance and surveillance activities, fuel handling, experiments, procedures, emergency preparedness, and safeguards and security. Other inspections, beyond those required by the routine inspection program, are completed as needed (e.g., in response to an event).

Those reactors that are shut down but are not being decommissioned have an abbreviated inspection program completed triennially. NRC inspects decommissioning reactors to verify their safe condition and the safe conduct of dismantlement and decontamination.

V. Emergency Preparedness

Emergency planning considerations for RTRs and power reactors are similar. The substantially smaller accident source term of the currently-licensed RTRs results in a significant reduction of potential radiological consequences of an accident. RTR licensees are required to identify the area for which emergency planning is performed. This area is defined as an Emergency Planning Zone (EPZ). The EPZ[27] comprises the area defined by emergencies that present potential radiological consequences that can result in off-site plume exposures that exceed 10 milliSieverts (mSv) [1 rem] deep dose or 50 mSv [5 rem] to the thyroid. For all currently licensed RTRs, the EPZ boundaries are established well within an area under the control of the licensee. For example, for 26 of the RTRs (those less than 2 MW_t), the EPZ is the operational boundary[28] and the EPZ of the highest thermal power NRC-licensed RTR (20 MW_t) is a 400 meter radius surrounding the facility.

RTRs also carry out an emergency classification scheme equivalent to that used at power reactors. The four emergency classes (notification of unusual event, alert, site area emergency, and general emergency) defined in 10 CFR Part 50, Appendix E, "Emergency planning and preparedness for production and utilization facilities," are used to classify RTR emergencies. However, the classification of general emergency is not used in emergency plans for any of the RTRs licensed because the postulated radiological consequences of the MHA at each RTR

[27] ANSI/ANS-15.16-2008, "Emergency Planning for Research Reactors"
[28] The operational boundary is the area within the site boundary, such as the reactor building, where the chief administrator has direct authority over all activities.

does not meet the criteria requiring a general emergency[29] declaration at the site boundary[30] (offsite).

VI. International Assessment of RTRs of the Fukushima Dai-ichi Accident

The NRC RTR staff has participated with our international partners in developing an understanding of the lessons learned from the Fukushima accident. This included work with both the Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA) on the development of international assessment guidance which included the application of a graded approach where appropriate. This work was deemed necessary because the stress test method initially used in Western European post-Fukushima assessments was designed specifically for power reactors and was applied to some of the European research reactors. Based on presentations at technical meetings from countries that applied the method to their research reactors, the stress test method was found to be much more beneficial in providing safety insight for the higher powered research reactors than for those of lower power. Application of the stress test method did not include grading of the assessment of the lower powered research reactors which resulted in the application of significant assessment resources that returned a diminished safety benefit.[31]

The NRC did not conduct stress tests for either power or non-power reactors. Instead, for power reactors, the staff followed the recommendations of the Near-Term Task Force, which are substantially addressing the same issues. For RTRs, the staff developed and followed the NRC assessment guidance for facilities other than power reactors. This guidance was compared to IAEA's Safety Report Series No. 80, "Safety Reassessment for Research Reactors in Light of the Accident at the Fukushima Dai-ichi Nuclear Power Plant," following issuance and found to be generally consistent.

International meetings, both at the NEA and IAEA, have been held to present the results of Member State reassessment of research reactor safety. Most countries have not fully implemented the draft guidance provided in IAEA Safety Report Series No. 80, opting only to revisit the seismic analyses of record for their research reactors. A few countries (mostly Western European) have met the intent of the IAEA guidance. NRC staff's assessment to date is that U.S. RTRs, while not fully compliant with IAEA Safety Report Series No. 80, do meet the intent of the document.

VII. Pre-Existing RTR Assessment Information Useful to the Assessment of the Fukushima Accident

Because of the terrorist attacks on September 11, 2001, the security at NRC-licensed facilities received increased attention. On September 28, 2001, the NRC Chairman directed NRC staff to undertake a thorough review of NRC's safeguards and security programs. The focus of this review was to examine basic assumptions underlying the regulatory framework and to find any

[29] 10 CFR Part 50 Appendix E
[30] The site boundary is that boundary, not necessarily having restrictive barriers, surrounding the operations boundary wherein the reactor administrator may directly initiate emergency activities.
[31] European Research Reactor Conference 2012.

necessary changes to optimize NRC, licensee, local, State, and Federal response capabilities. In carrying out this directive, NRC staff completed an integrated and comprehensive security assessment to determine if physical security improvements were warranted at the various types of NRC-licensed facilities, including RTRs.

The 2006 RTR security assessments focused on theft and sabotage scenarios at each RTR. These scenarios were chosen as they were viewed as having the potential to adversely affect public health and safety. The postulated consequences of these scenarios were compared to NRC-established radiological consequence screening criterion to determine if adequate protection of public health and safety is maintained. NRC staff employed a three-phase, assessment process that the Commission had approved.[32] The 2006 security assessments are not publically available because they contain sensitive security-related information.

The post-9/11 RTR security assessments concluded that the security posture at NRC-licensed RTRs remains adequate to promote the common defense and security and protect public health and safety. The extension of security assessment conclusions related to the analysis of sabotage scenarios can, at a minimum, offer useful insights to assist in the assessment of the consequences associated with a beyond-design-basis external event because both may result in severe damage states to the facility that may exceed those assumed in RTR's design and accident analysis. The post-9/11 security assessments considered the potential radiological consequences for the various sabotage scenarios. The supporting analyses of the radiological consequences demonstrated that RTR sabotage scenarios would result in doses to the public that are fractions of the 10 CFR Part 100 reactor siting dose criteria.[33] NRC staff considered the sabotage event assessment and its conclusions as part of the beyond-design-basis external event assessment and determined that the sabotage assessment conclusions could not be extended to the beyond-design-basis external event case without additional site specific assessment.

VIII. Prompt Post-Fukushima Assessments

In the days immediately following the Fukushima accident, NRC staff collected available information related to accident initiation, progression, and consequences. The NRC RTR staff used this information to inform a prompt assessment of the safety of NRC-licensed RTRs. The goal of the assessment was to determine if insights gained from the accident revealed any conditions or lessons learned that would call into question the safety of the NRC-licensed RTRs or reveal the need for immediate regulatory action. NRC staff conclusions were based on the available information and engineering judgment. The areas considered by NRC staff during the prompt assessment included:

- natural events

[32] SRM SECY-04-0222, dated January 19, 2005

[33] Per cited regulations, an individual located at any point on the exclusion area boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

- electrical power
- decay heat removal
- spent fuel
- combustible gas control
- reactor containment/confinement.

Natural Events

The licensing process for RTRs requires that the applicant describe and discuss the geographical, geological, seismological, hydrological, and meteorological characteristics of the site. This information must be presented in sufficient detail to allow NRC staff to compare the site's potential natural hazards to the applicant's proposed facility design. This allows NRC staff to decide on the acceptability of the proposed design at that site. To support such a conclusion, the design must provide reasonable assurance that structures, systems and components will remain capable of performing safety functions during and after postulated natural events.

After the Fukushima Dai-ichi accident, the staff reviewed the natural event descriptions and discussions each RTR licensee supplied in licensing documentation. NRC staff performed the first of these assessments the day of the accident. The staff specifically assessed the designs of those RTRs in Pacific coastal states in light of the predicted tsunami forecasted to impact the Pacific coast of the United States. The staff concluded that none of the five RTRs in Pacific States were vulnerable to the tsunami reaching the locations of the facilities.

In the weeks after the accident, NRC staff started a broader assessment of natural events originally considered in the design bases for the NRC-licensed RTRs. A specific focus was placed on the seismic and flooding events. NRC staff reached the following general conclusion related to the current design and siting of the NRC-licensed RTRs:

- The treatment of natural events for each NRC-licensed RTR is reviewed at the time of licensing and considers the average and extreme (historical worst case) values as well as predictive potentials (where appropriate and available) in the assessment of the potential hazards presented by a specific natural event.

- Because of their low power, NRC-licensed RTRs do not rely on large sources of water for makeup and heat sinks which allows siting of RTR facilities a significant distance from potential sources of flooding.

- Each research reactor licensee has demonstrated through various analyses that the radiological consequences associated with the maximum predicted seismic event at the facility site, using information available at the time of licensing, meets accident analysis acceptance criteria which have been based on highly conservative public and occupational dose limits. (RTRs do not update their seismic analyses with the latest seismic hazards information unless undergoing relicensing)

- Similar to power reactors, the one test reactor currently licensed by the NRC must meet 10 CFR Part 100 seismic requirements and radiological limits.

Electrical Power

Given the Fukushima scenario, NRC staff included a review of an extended loss of electrical power on the safety of NRC-licensed RTRs as part of their immediate assessment. NRC staff reached the following conclusions related to the NRC-licensed RTRs' reliance on electrical power and sensitivity to its loss:

- Most RTRs have some level of emergency power, typically to power area radiation monitors, evacuation alarms and lighting, and security systems. None require electrical power (normal or emergency) to safely shut down the reactor.
- 28-out-of-31 RTRs are air-coolable and thus do not require electric power for decay heat removal.
- 2-out-of-31 RTRs may require ac power to replenish the reactor coolant inventory lost because of seismically-induced failure of the reactor coolant system boundary (pool or tank) or boil-off.

Decay Heat Removal

At Fukushima it was ultimately the inability to remove the decay heat[34] from the reactor cores that resulted in catastrophic failures. The generated decay heat is not unique to power reactors and is also generated in the cores of RTRs. The major differences between power reactors and RTRs are the magnitude of their maximum operating power, the duration of operation, and the smaller quantities of spent or irradiated fuel onsite. These qualities equate to a significantly smaller fission product inventory and a significantly reduced decay heat generation rate at RTRs. These along with other factors make RTRs much less susceptible to core damage from overheating when compared to power reactors and generally eliminate the need for highly complex, diverse, and redundant active decay heat removal systems typical of those found at power reactors. For example, for the NRC-licensed RTRs that are less than 2 MW_t, decay heat can be adequately removed through air cooling of the core.[35]

Given the Fukushima scenario, NRC staff included a review of the decay heat removal capability as part of the immediate assessment. NRC staff reached the following conclusions related to the decay heat removal at NRC-licensed RTRs:

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[34] The majority of the decay heat generated is attributable to the decay of the radioactive fission products. After reactor shutdown, decay heat is about 6.5 percent of the previous operating power and decreases over time; 1.5 percent after 1 hour, 0.4 percent after 24 hours, and 0.2 percent after one week. Ref. DOE-HDBK-1019, "DOE Fundamentals Handbook – Nuclear Physics and Reactor Theory," January 1993.

[35] S. Hawley and R. Kathren, Pacific Northwest Laboratory, 1982. "NUREG/CR-2387, Credible Accident Analyses for TRIGA and TRIGA Fueled Reactors."

- Natural convection of primary coolant provides adequate decay heat removal for all RTR designs in the short-term (0.5 to 2.5 hours).

- The one test reactor licensed by the NRC requires active systems for adequate long-term decay heat removal.

- In cases of loss of coolant scenarios, air cooling is sufficient to remove decay heat for all RTRs with maximum licensed power levels of less than or equal to 2 MW_t (28 of the 31 RTRs).

- Loss of coolant scenarios in the greater than 2 megawatt reactors (3 of the 31 RTRs) rely on designs that:

  – Maintain floodable volume that maintains the core covered with water at those facilities.

  – Provide the capability to spray the core with diverse sources of water via active (pumped) and/or passive (gravity drain) means.

  – Are equipped with emergency power systems or batteries sufficient to power cooling and makeup systems for the time required following reactor shutdown.

Spent Fuel Pool Cooling

Given the public interest in spent fuel pools following the Fukushima accident, NRC staff included a review of the storage of irradiated and spent fuel at RTRs. NRC staff reached the following conclusion:

- All RTRs have very small inventories of spent or irradiated fuel.

- At all but the three largest RTRs, spent fuel is not routinely discharged due to low power and limited operational duration and frequency (practically speaking, the majority of the less than 2 MW_t RTR reactor cores can be considered lifetime cores).

- The Department of Energy (DOE) recovers spent and/or unwanted irradiated fuel from 28 of 31 RTRs owned by government or academic institutions which includes the largest generators of spent fuel preventing the accumulation of large on-site spent fuel inventories.

- Typically, spent, irradiated and excess fuel is stored dry, in the reactor pools, or in dedicated spent fuel pools when provided by design.

- In no case does the storage of spent or irradiated RTR fuel require active cooling to remove adequate decay heat.

Combustible Gas Control

Significant damage resulted to the primary and secondary containment structures at Fukushima Dai-ichi due to ineffective control of hydrogen. Given the Fukushima scenario, NRC staff included a review of hydrogen generation and control at RTRs. NRC staff reached the following conclusions:

- Low-power RTRs (those that are licensed to operate at less than or equal to 2 MW_t) do not generate sufficient quantities of hydrogen from the radiolytic decomposition of water to reach combustible or explosive concentrations of hydrogen given the volume and mixing of the hydrogen with the reactor building atmosphere.

- RTRs do not inject hydrogen into the primary coolant to scavenge dissolved oxygen.

- The higher-powered RTRs either prevent the formation of combustible or explosive concentrations of hydrogen through the use of a dedicated hydrogen control system or have demonstrated by analysis that the formation of combustible or explosive concentrations are not possible under normal operating and accident conditions.

- Concerning the formation of hydrogen from metal water reactions during accident conditions:

  – The amount of reactive metal present in a RTR core is several thousand times less than the reactive metal contained in a power reactor core.

  – Only one RTR uses zirconium cladding and this reactor does not generate sufficient decay heat to reach the elevated cladding temperature necessary to initiate the zirconium-water reaction.

  – RTRs with stainless steel fuel cladding also lack sufficient decay heat to reach the elevated cladding necessary to initiate a metal-water reaction.

  – An aluminum-water reaction at an RTR with aluminum fuel cladding is also extremely unlikely due to the necessity to establish precise physical conditions related to the form of the aluminum (i.e., small droplets of molten aluminum) to initiate the reaction.

Reactor Containment/Confinement

As discussed above, the Fukushima Dai-ichi accident resulted in catastrophic failure of both the primary and secondary containments at some of the affected units. These failures did not occur directly from either the earthquake or the tsunami but only because of the inability to remove decay heat at a sufficient rate. This led to the eventual failure of all barriers intended to prevent the release of radioactive materials.

All but three of the RTRs are designed with confinement structures that are not pressure tight structures. They typically employ a ventilation system to maintain the reactor building at a slight negative pressure and discharge through a controlled pathway with a filtered and elevated release point.

Three research reactors use containment buildings that provide a controlled leakage boundary. The containment structures are equipped with both under and over pressure protection thereby preventing the failure of the containment building from excessive differential pressures.

Given the insights gained from the Fukushima accident, NRC staff reviewed the containment and confinement structures used at RTRs to prevent or control the release of radioactive materials to the environment. NRC staff reached the following conclusions:

- NRC-licensed RTRs operate at very low power, low temperature (well below the boiling point of water at atmospheric pressure), and low pressure (typically at or near atmospheric pressure) and as such the energy that must be controlled or dissipated during an accident is very low and orders of magnitude less than at a power reactor.

- Energetic releases from the reactor or reactor coolant systems to the containment or confinement are not expected to challenge the design limits during normal operating, transient, or accident conditions due to low operating temperatures and pressures, large containment/confinement volumes, and relatively low decay heat generation rates.

Prompt Assessment Conclusions

The RTR prompt assessment was completed within weeks of the Fukushima Dai-ichi accident. As discussed above, NRC staff concluded that there were no safety concerns revealed by the accident for which immediate actions were necessary, nor was any new information revealed that would contradict or invalidate assumptions used in the safety basis of any of the RTRs licensed by the NRC.

IX. Assessment of Near-Term Task Force Report Recommendations to RTRs

In days after the Fukushima Dai-ichi accident, the Commission directed NRC staff to review methodically the NRC's processes and regulations to determine whether the NRC should make improvements to its regulatory system and to make recommendations to the Commission regarding policy direction. In response to the Commission's direction, the Executive Director for Operations established the Near-Term Task Force (NTTF) to conduct a near-term evaluation of the need for agency action.

On July 12, 2011, the NTTF issued their report, "Recommendations for Enhancing Reactor Safety in the 21st Century" (ADAMS Accession No. ML111861807). The report has 12 recommendations. The focus of the NTTF's review was specific to power reactors, but their recommendations can be generally considered for any type of licensed facility. NRC staff developed a review process that provided general guidance for the review of the 12 NTTF recommendations, external events, and other Fukushima lessons learned for licensed facilities other than power reactors.

In assessing the applicability of lessons learned from the Fukushima Dai-ichi accident to RTRs, NRC staff chose to first group the RTRs based on their licensed thermal power level, because a given facility's susceptibility to core damage under conditions similar to those encountered during the Fukushima Dai-ichi accident correlate directly to thermal power and the decay heat generation rate. This resulted in two review categories. Category 1 included research reactors with a licensed thermal power level of less than 2 MW_t, which included 26 of 31 NRC-licensed research reactors. Category 2 included the remaining four research reactors, and the one test reactor, all of which were licensed for thermal power levels of 2 MW_t or greater.

Using a 2 MW_t threshold for categorizing RTRs was not arbitrary. Some analyses[36] exist that demonstrate that air cooling is sufficient to remove decay heat from research reactors licensed for operation at less than 2 MW_t after a complete loss of coolant (dry core). One analysis of research reactor plate-type fuel[37] (commonly referred to as MTR-type fuel) concludes that sufficient decay heat removal by air cooling alone exist for reactors up to 3 MW_t. More recent facility-specific accident and thermal hydraulic analyses conducted in support of highly-enriched uranium (HEU) to low enriched uranium (LEU) conversions, or renewed operating licenses at research reactors of less than 2 MW_t, have confirmed those conclusions.

Recent security-related studies and assessments also contribute to the justification of the 2 MW_t categorization criteria for RTRs. These studies and assessments did not focus on fuel failure due to inadequate decay heat removal but rather on intentional damage or destruction of the fuel. Of primary concern was the magnitude of the potential radiological consequences resulting from a sabotage event. The sabotage studies conducted as part of the regulatory basis for the rulemaking that added 10 CFR 73.60(f), "Additional requirements for physical protection at non-power reactors," provided the initial justification for greater security requirements to protect against sabotage at RTRs rated at thermal power greater than or equal to 2 MW_t. The sabotage study results were later expanded upon by the 2006 security assessments conducted in response to the terrorist attacks of September 11, 2001. These subsequent RTR security assessments reviewed the adequacy of the existing security framework for the protection of nuclear materials from theft and sabotage in light of the new threat. The review concluded that the existing protections against sabotage remained adequate.

Assessment of External Events on RTRs

NRC staff assessed the various external events listed in the staff guidance to determine if an assumed external event exceeding the magnitude of the external events analyzed during the licensing process would result in more severe radiological consequences than those postulated in accident analyses. The external events analyzed during the licensing of each RTR were based on the average and extreme historical values and predictive potentials for the specific type of event and location of the facility.

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[36] S. Hawley and R. Kathren, Pacific Northwest Laboratory, 1982. "NUREG/CR-2387, Credible Accident Analyses for TRIGA and TRIGA Fueled Reactors."
[37] C. Webster. "Water-Loss Tests in Water-Cooled and Moderated Research Reactors." Nuclear Safety, Volume 8, Number 6, November-December 1967.

RTRs less than 2 MW_t (Category 1 Research Reactors)

NRC staff found that the risk of a significant release of radioactive material because of external events exceeding the severity of the external events considered during the licensing is very low for the research reactors licensed for operation at less than 2 MW_t. As discussed above, the low thermal power rating of these research reactors results in a low decay heat generation rate such that air cooling of the fuel would prevent overheating of the fuel cladding even if the external event in question causes or happens concurrently with the complete loss of coolant, the loss of electrical power, and the loss of all active decay heat removal systems.

The table below summarizes the external events reviewed by NRC staff, the impact on the facility and a summary of the basis for no additional action being needed for this class of RTRs.

Effect of External Events on Category 1 Research Reactors

External Event	**Potential Effect on the Facility**	**Assessment**
Flood	• Loss of all electrical power and any active heat removal systems. • Potential damage to facility.	• Active decay heat removal systems and electrical power are not needed. Decay heat is removed by natural convection of pool water or by air cooling if pool inventory is lost. • The operation of these facilities is not reliant on large bodies of water and therefore not typically sited directly adjacent to them (with one exception)[38] thereby reducing the susceptibility to flooding.
Seismic	• Loss of all electrical power and any active heat removal systems. • Challenge to reactor structures including confinements or containments. • Physical damage to reactor components (e.g., pool integrity, passive cooling features or structural components).	• Active decay heat removal systems and electrical power are not needed. Decay heat removed by natural convection of pool water or by air cooling if pool inventory is lost. • Facilities in this category are insensitive to changes in the seismic hazard since they are air coolable in the event of a seismic induced loss-of-coolant accident. • Building code bases is to prevent building collapse.

[38] A 100 watt critical assembly is located directly adjacent to a river and has historically experienced flooding that resulted in minimal impact to the facility. While water would damage the facility, it will not have a detrimental effect on decay heat removal which can be achieved by air cooling alone.

External Event	Potential Effect on the Facility	Assessment
		• Loss of air cooling from debris obstruction highly unlikely. • Fuel cladding remains intact. • No radiological release expected.
High Wind, Tornado, and Missiles	• Loss of all electrical power and active heat removal systems. • Challenge to reactor structures including confinements or containments.	• Below grade design or above grade reinforced concrete biological shield prevents impact damage to fuel. • Bounded by seismic assessment above.
Lightning	• Loss of all electrical power and active heat removal systems.	• Bounded by flood and seismic assessment above.
Snow and ice loads	• Loss of all electrical power and active heat removal systems. • Challenge to reactor structures including confinements/containments. • Physical damage to reactor components (e.g., pool integrity, passive cooling features or structural components).	• Bounded by flood and seismic assessment above.
Drought and Temperature Extremes	None	• Bounded by flood and seismic assessment above.
Fire	• Loss of all electrical power and active heat removal systems. • Challenge to reactor structures including confinements or containments. • Physical damage to reactor components (e.g., pool integrity, passive cooling features or structural components).	• Low combustible loading and compliance with fire codes at the reactor facility. • Below grade design or above grade reinforced concrete biological shield prevents direct heating of pool liner or the fuel. • Bounded by flood and seismic assessment above.
Loss of Power	• Loss of all electrical power and active heat removal systems. • Challenge to reactor	• Bounded by flood and seismic assessment above.

External Event	Potential Effect on the Facility	Assessment
	confinement due to loss of ventilation systems.	
Loss of Heat Sink	• Require a reactor shutdown at most RTRs to prevent exceeding temperature limits.	• Bounded by flood and seismic assessment above.

RTRs of Greater Than or Equal to 2 MW_t (Category 2 Research and Test Reactors)

Air cooling for the Category 2 RTRs can no longer be assumed to provide adequate decay heat removal based solely on the RTRs maximum licensed thermal power. The analysis referenced in footnote 37 concluded through loss of coolant testing that a research reactor operating at 3 MW_t for 1 week could lose coolant via a tank rupture without experiencing fuel clad failure. As a matter of conservatism, the NRC has limited consideration of air cooling to less than 2 MW_t. More recent testing referenced in footnote 36 also confirmed that a TRIGA research reactor operating at 2 MW_t when subjected to an instantaneous loss of coolant via a beam tube, piping, or tank rupture would not experience fuel cladding failure. Given the potential for increased decay heat generation at reactors licensed for a thermal power of 2 MW_t or greater requires reliance on specific operating conditions such as the availability of reactor coolant, a heat sink, and electrical power become more important.

The decay heat studies referenced previously in this paper assumed the reactor that operated continuously at its maximum licensed power of 2 MW_t. If one considers a more realistic and representative power history for a specific reactor, some of the Category 2 research reactors would respond similarly to that of the research reactors in Category 1 to severe external events. Adopting the concept used in the 2006 RTR security assessments of calculating a facility effective power (FEP) will represent a more accurate approximation of the actual decay heat that must be removed to prevent fuel failure. The FEP for the two 2 MW_t facilities produce less decay heat than for a 1 MW_t research reactor (0.12 MW_t and 0.81 MW_t). Given the fractional use of their maximum licensed power over the life of these facilities, it can be demonstrated that decay heat can be removed adequately by natural convection of pool water or by air cooling in the event that coolant is lost without need for active decay heat removal systems and electrical power. Because of the similarities between the 2 MW_t Category 2 reactors and the Category 1 reactors, the radiological consequences postulated in their MHAs would bound external events. Therefore, NRC staff considers the analysis summarized in the above table for RTR less that 2MW_t to be applicable to the 2 MW_t reactors.

For the three remaining reactors in Category 2, FEPs exceeded 2 MW_t indicating that air cooling would not be adequate to prevent fuel cladding failure. NRC staff's focus on external event assessment of Category 2 RTRs will be on the 6 and 10 MW_t, research reactors and the 20 MW_t test reactor only.

Research Reactors Licensed for Greater Than 2 MW_t

There are two high powered research reactors in Category 2. They include the University of Missouri at Columbia (MURR) and Massachusetts Institute of Technology (MITR) reactors. Both of these reactors are tank type reactors capable of removing adequate decay heat by the natural convection flow of the reactor coolant following a severe external event even if that event results in the loss of all electrical power and active decay heat removal systems. In this case, decay heat is not sufficient (given the passive heat sink) to raise the temperature of the water above bulk boiling. Therefore, for this scenario, there is not a near-term need to replenish the water around the reactor fuel lost by evaporation. It is only when the initiating external event also causes (or occurs concurrently with) a loss of primary coolant, a condition which would require the failure of the core tank and reactor pool integrity, do the conditions exist that result in inadequate decay heat removal.

A type of severe external event that could result in conditions where there is a loss of coolant accident (LOCA) and concurrent loss of ac power is a seismic event. A seismic event would be required to exceed significantly the predictive potential for seismic activity in the areas where these reactors are located to cause the failure of the core tanks and the reactor pools. For example, the MITR is in an area where the USGS's 2014 seismic hazard data for the reactor's location predicts a maximum peak ground acceleration of 0.16 g. A seismic analysis referenced in the MITR Safety Analysis Report for the core tank demonstrates that it is capable of withstanding, at yield stresses, static forces corresponding to 5.1 g horizontal simultaneously with 3.4 g vertically. Based on this analysis, the licensee concluded that the seismically-induced loss-of-coolant accident is not credible. Considering the similarity between the designs of the MIT and MURR, it would be reasonable to assume that both will have seismic margins; however, the magnitude of that margin will vary depending on local seismic hazards. The staff plans to confirm that the seismic margins for these research reactors are such that a LOCA induced from a seismic event is not credible. If necessary, the staff may look at the time needed before core damage and inform the size and location of the break through structural evaluations. If the seismic margins are such that additional regulatory steps are necessary, the staff will follow the appropriate regulatory process to mitigate the potential for a seismically-induced LOCA concurrent with loss of ac power.

NRC staff is also assessing the capability of other external events (e.g., missiles from high winds) resulting in a LOCA and concurrent loss of electrical power. Both of the research reactors in this category have containments and the designs are such that the reactor core is protected by above grade reinforced concrete biological shield that provide an additional barrier (in addition to the containment) that minimize the potential for impact damage to the pool liner, core tanks and fuel. As part of the seismic evaluation above, NRC staff will confirm that external events other than seismic events are not credible causes of a LOCA concurrent with a loss of ac power.

For both the seismic evaluation assessment and the assessment of the capability of other external events to cause a LOCA and concurrent loss of ac power NRC staff will consider the results of the previous security assessments before other regulatory actions, if any, are considered. As part of their assessment of these events, NRC staff plans to gather the appropriate structural expertise, supplemented by NRC staff members who have performed assessments of power reactor responses to NRC's Order EA-12-049, to inform its assessment on whether additional regulatory action is needed. The table below summarizes the external events reviewed by NRC staff, the impact on the facility, and the staff's assessment.

Effect of External Events on MURR and the MITR Research Reactors

External Event	Potential Effect on the Facility	Assessment
Flood	• Loss of all electrical power and any active heat removal systems. • Challenge to facility equipment and reactor structures including confinements or containments.	• Decay heat is adequately removed by natural convection of pool water. Inundation by flood waters will undoubtedly damage facility equipment but it is not expected to interrupt decay heat removal through natural convection.
Seismic	• Loss of all electrical power and any active heat removal systems. • Challenge to reactor structures including confinements or containments. • Physical damage to reactor components (e.g., pool integrity, passive cooling features or structural components).	• The building codes are established to prevent building collapse and would limit the likelihood of catastrophic failure of structures. • The reactors in this class have better seismic capability than the Category 1 research reactors. • The research reactor decay heat is adequately removed by natural convection of the pool water. Active decay heat removal systems and electrical power are not needed. • If the seismic event either causes or occurs concurrently with the loss of coolant, these research reactors may require support from portable equipment if installed equipment cannot be promptly restored. NRC staff to continue assessment.
High Wind, Tornado, and Missiles	• Loss of all electrical power and active heat removal systems. • Challenge to reactor structures including confinements or containments.	• Above grade reinforced concrete biological shield and containment wall may prevent impact damage to the reactor pool and fuel. • NRC staff to continue assessment to ensure that there is not an event-specific vulnerability.

External Event	Potential Effect on the Facility	Assessment
Lightning	• Loss of all electrical power and active heat removal systems.	• Decay heat is adequately removed by natural convection of pool water. Active decay heat removal systems and electrical power are not needed.
Snow and ice loads	• Loss of all electrical power and active heat removal systems. • Challenge to reactor structures including confinements or containments.	• Bounded by seismic and high winds assessment. • NRC staff will continue assessment to ensure there is not an event-specific vulnerability.
Drought and Temperature Extremes	• It may be necessary to shut down the reactors based on an inability to continue to meet Technical Specifications.	• Active decay heat removal systems and electrical power are not needed.
Fire	• Loss of all electrical power and active heat removal systems. • Challenge to reactor structures including confinements or containments.	• Low combustible loading and compliance with fire codes at the reactor facility. • Fire-induced LOCA highly unlikely. In the event of a fire the biological shield and water in reactor core tank and reactor pool will cool these components such that fire-induced LOCA is not credible.
Loss of Power	• Loss of all electrical power and active heat removal systems.	• Active decay heat removal systems and electrical power are not need.
Loss of Heat Sink	• Require a reactor shutdown at most RTRs to prevent exceeding temperature limits.	• Active decay heat removal systems and electrical power are not needed.

20 MW_t Test Reactor

The National Institute of Standards and Technology (NBSR) test reactor is protected initially from fuel cladding failure via a passive coolant makeup system combined with natural convection cooling following the loss of active decay heat removal capability. Specifically, the inner reserve tank will make up the inventory boiled off during the first half hour. The emergency tank will continue to replenish the inventory for an additional 2 hours. As the inventory of coolant contained in the passive makeup system becomes depleted, the pool coolant level boils down until the fuel becomes uncovered and fuel temperatures start to increase to a point where fuel cladding can fail if the facility's light water makeup water source to the core or electrical power and a means of decay heat removal are not restored to operation or otherwise provided via portable equipment in a timely manner. This portable equipment can allow domestic light water to be supplied to the reactor following installation of a spool-piece. Emergency power to the active decay heat removal system can be supplied by the 125 volt direct current station batteries for a limited time or from one of the two on-site emergency diesel generators for a longer period of time. In the event of a LOCA, the heavy water can be collected in sumps and then pumped back into the reactor for decay heat removal; however, this requires ac power. The test reactor fuel cladding is vulnerable to failure to beyond-design-basis seismic events that potentially result in an extended loss of electrical power, active decay heat removal systems, or coolant inventory makeup capability.

The radiological consequences resulting from a severe external event may exceed those assumed in the MHA but would not exceed 10 CFR Part 100 siting criteria used in licensing the facility. This has been confirmed by the post-9/11 security assessment of sabotage scenarios which assumed massive damage states to the facility. Because of the malicious intent and the extreme assumptions of facility damage used in the sabotage assessment, the postulated radiological consequences from the worst case sabotage event are expected to bound the postulated radiological consequences of all external events. The radiological consequences predicted by the worst case sabotage event analysis are a fraction of the 10 CFR Part 100 reactor siting dose criteria.

Nevertheless, NRC staff plans to review the seismic capability for this reactor to determine whether an extended loss of ac power (both offsite and loss of on-site emergency diesel generators) from a seismic event is credible. If the seismic assessment's results are such that additional mitigative capabilities are necessary, NRC staff will follow the appropriate regulatory process to mitigate the potential for a seismically-induced extended loss of ac power.

Flooding at the test reactor is not a likely external event. No major bodies of water are near to the facility and the facility is above the 500 year flood plain. Because no major sources of water are near the facility, there are no water control structures, such as dams, dikes or levees subject to seismically induced failure which could result in the flooding of the test reactor. Records show flooding from local intense precipitation (LIP) events is extremely rare. The flooding of the facility from a LIP event which causes the extended loss of all electrical power and active decay heat removal is also not of concern because such an event would not adversely impact the availability of heavy and light water makeup sources to the core. In the long-term, makeup water can be replenished though connections to the city water system, which do not require electrical power. NRC staff plans to continue its assessment of flooding events to determine if additional regulatory actions are needed to address this event.

NRC staff is also assessing the capability of other external events (e.g., missiles from high winds) causing an extended loss of all (offsite, onsite, and emergency sources) electrical power. As part of its assessment of these events NRC staff intends to gather the appropriate structural and flooding expertise, supplemented by NRC staff members who have assessed power reactor responses to NRC's Order EA-12-049, to inform its assessment on whether additional regulatory action is needed. The table below summarizes the external events reviewed by NRC staff, the effect on the facility and the NRC staff's assessment.

Effect of External Events on NBSR

External Event	Effect on the Facility	Assessment
Flood	• Loss of all electrical power and any active heat removal systems. • Challenge to facility equipment and reactor structures including confinements or containments.	• Passive coolant makeup system removes decay heat for 2.5 hours. • City-water backup can supplement makeup coolant inventory after 2.5 hours, if available. • Bounded by seismic results
Seismic	• Loss of all electrical power and any active heat removal systems. • Challenge to reactor structures including confinements or containments. • Physical damage to reactor components (e.g., pool integrity, passive cooling features or structural components).	• Test reactor meets 10 CFR Part 100 siting criteria. • The test reactor decay heat is adequately removed by natural convection of reactor vessel water for a minimum of 0.5 hours. An additional 2 hours of makeup water is provided via the emergency tank if it survives the seismic event. Continued adequate decay heat removal will require either a source of makeup water or AC power and active decay heat removal. • NRC staff will continue assessment to ensure there is not an event-specific vulnerability
High Wind, Tornado and Missiles	• Loss of all electrical power and active heat removal systems. • Challenge to reactor structures including confinements or containments.	• The confinement building and the biological shield provide barriers to wind driven missiles. • The confinement building and reinforced concrete biological shield may prevent impact damage to reactor vessel and fuel. Subject of additional assessment. • Test reactor decay heat is adequately removed by natural convection of pool water initially. Continued adequate decay heat removal will require either a source of makeup water or active decay heat removal systems for the long-term. • NRC staff will continue assessment to ensure there is not an event-specific vulnerability.
Lightning	• Loss of all electrical power and active heat removal	• Test reactor decay heat is adequately removed by natural

External Event	Effect on the Facility	Assessment
	systems.	convection of reactor vessel water initially. Continued adequate decay heat removal will require active decay heat removal systems in the long-term. • NRC staff will continue assessment to ensure there is not an event-specific vulnerability.
Snow and ice loads	• Loss of all electrical power and active heat removal systems. • Challenge to reactor structures including confinements or containments.	• Test reactor decay heat is adequately removed by natural convection of reactor vessel water initially. Continued adequate decay heat removal will require active decay heat removal systems in the long-term. • NRC staff will continue assessment to ensure there is not an event-specific vulnerability.
Drought/ Temperature Extremes	It may be necessary to shut down the reactors based on an inability to continue to meet Technical Specification.	• Test reactor decay heat is adequately removed by natural convection of pool water initially. Continued adequate decay heat removal will require active decay heat removal systems in the long-term. • Long-term coolant makeup requirements very minimal for this reactor because of low decay heat. • NRC staff will continue assessment to ensure there is not an event-specific vulnerability.
Fire	• Loss of all electrical power and active heat removal systems. • Challenge to reactor structures including confinements or containments.	• Low combustible loading and compliance with fire codes at the reactor facility. • Above grade reinforced concrete biological shield prevents direct heating of the reactor vessel and fuel. • Test reactor decay heat is adequately removed by natural convection of reactor vessel water initially. Continued adequate decay heat removal will require active decay heat removal systems in the

External Event	Effect on the Facility	Assessment
		long-term. • Staff will continue assessment to ensure there is not an event-specific vulnerability.
Loss of Power	• Loss of all electrical power and active heat removal systems. • Challenge to reactor confinement.	• Test reactor decay heat is adequately removed by natural convection of reactor vessel water initially. Continued adequate decay heat removal will require active decay heat removal systems in the long-term. • NRC staff will continue assessment to ensure there is not an event-specific vulnerability
Loss of Heat Sink	• Require a reactor shutdown at most RTRs to prevent exceeding temperature limits.	• Test reactor decay heat is adequately removed by natural convection of reactor vessel water initially. Continued adequate decay heat removal will require active decay heat removal systems in the long-term. • NRC staff will continue assessment to ensure there is not an event-specific vulnerability.

Assessment of NTTF Recommendations

Assessment Conclusions of NTTF Recommendations for RTRs <2 MW_t

Regarding the NTTF recommendations associated with ensuring protection and enhancing mitigation, the research reactors in Category 1 are resilient to cladding failure due to decay heat induced overheating. They do not require electric power, active decay heat removal systems, or the presence of reactor coolant to adequately remove decay heat. They are small facilities that operate infrequently and at low power and as such present minimal radiological hazards to public health and safety. Confinement or containment structures are not challenged by energetic releases from primary or secondary cooling systems or by hydrogen generation from metal-water reactions during accidents. The research reactors in this category do not generate significant quantities of spent fuel and the minimal quantities that exist can be adequately cooled by air.

Regarding the NTTF recommendations associated with emergency preparedness, NRC-licensed research reactors are not collocated with other reactors. Emergency preparedness response would be anticipated to be uncomplicated due to their simplistic design, low decay heat and small source terms. EPZs are small. Facility recovery and reentry following an accident would likely be successful through the implementation of uncomplicated manual actions and prudent health physics precautions and controls.

Assessment Conclusions of NTTF Recommendations for RTRs
Greater Than or Equal to 2 MW_t

Regarding the NTTF recommendations associated with ensuring protection and enhancing mitigation, the three highest power Category 2 reactors (MURR, MITR, and NBSR) are particularly sensitive to the availability of reactor coolant. The early loss of reactor coolant can result in failure of the fuel cladding and subsequent radiological release unless reactor coolant makeup can be provided from installed facility equipment or from portable external sources. If coolant remains available, all three of these reactors will initially be provided with adequate decay heat removal via natural circulation cooling. However, if there is an extended loss of electrical power to operate the active decay heat removal systems or damage to the active decay heat removal system that prevents its use, the one test reactor could require the recovery of those systems or be provided with coolant make up from portable external sources to prevent fuel cladding failure. Therefore, the NTTF recommendations related to these scenarios could prove beneficial, under rare circumstances, in preventing fuel cladding failure at the test reactor. As discussed in the section above regarding external events, NRC staff is continuing its assessment of the need for protective or mitigative strategies for these reactors. If NRC staff concludes that additional protective or mitigative strategies are appropriate, NRC staff will use existing regulatory processes to carry them out.

Regarding the NTTF recommendations associated with enhancing mitigation for containments, the Category 2 reactors present an increased radiological hazard due to decay heat generation rates as well as a proportional increase in fission product inventory. Category 2 reactor confinement or containment structures are not challenged due to the absence of any significant energetic releases from primary or secondary cooling systems or by the generation of combustible or explosive concentrations of hydrogen from metal-water reactions during accidents.

Regarding the NTTF recommendation associated with enhancing mitigation for spent fuel pools, the RTRs in this category generate small quantities of spent fuel that can be adequately cooled by air shortly after discharge from the core. Additionally, the DOE routinely recovers spent fuel from these facilities preventing the long-term accumulation of spent fuel on site.

Regarding the NTTF recommendations associated with strengthening emergency preparedness, NRC-licensed research reactors are not collocated with other reactors. Emergency preparedness response would be anticipated to be uncomplicated due to their relative simplistic design, low decay heat and small source terms. EPZs are small and do not extend beyond the owner controlled area. Facility recovery and reentry following an accident would likely be successful through the implementation of uncomplicated manual actions and prudent health physics precautions and controls.

X. Conclusions and Recommendations

Research Reactors of Less than 2 MW_t (Category 1)

The reactors in this category represent most of the NRC-licensed research reactors (26 of 31) with a maximum licensed thermal power of 1.1 MW_t. They do not rely on the availability of electrical power, active decay removal systems, or coolant inventory to adequately remove decay heat and prevent fuel clad failure. Even if a loss of coolant results from, or occurs concurrently with, an external event, sufficient decay heat can be removed via air cooling of the core. This resiliency is attributable to short duration and low power operation typical of this category of reactor and results in a minimal fission product inventory and decay heat generation.

RTRs of Greater Than or Equal to 2 MW_t (Category 2)

The reactors in this category represent the highest powered of the NRC-licensed RTRs (5 of 31) with licensed thermal power levels from 2 to 20 MW_t. NRC staff has not found other actions are needed for the two 2 MW_t research reactors in Category 2.

The three highest powered RTRs in this category do rely on the presence of a coolant inventory to adequately remove decay heat. The research reactors in this category are not reliant on active decay heat removal systems and electrical power to prevent fuel cladding failure; however, the test reactor is. After the depletion of the coolant from the test reactor's passive coolant makeup system and heavy water storage tank, the test reactor would require electrical power and the active decay heat removal system or additional sources of coolant.

As discussed above, NRC staff is performing additional assessments related to the peak ground acceleration required to result in the failure of the primary coolant system integrity for the two high power research reactors. As part of this assessment NRC staff will also assess the capability of other external events (e.g., missiles from high winds) causing an issue with primary coolant integrity. The goal of this assessment is to determine if it is necessary to include additional features that would prevent or mitigate the loss of primary coolant. If it can be shown that pool failure is not credible under these circumstances, then NRC staff is unlikely to pursue additional actions for these research reactors.

It is important to note that the test reactor was sited in accordance with 10 CFR Part 100. As part of the licensing process, the test reactor applicant provided evidence with the analysis of postulated accidents, including the MHA, that the 10 CFR Part 100 reactor siting dose criteria were met. The 2006 security assessments performed in response to the terrorist attacks of September 11, 2001, used the 10 CFR Part 100 reactor siting dose criteria as criteria to justify that no additional security-related regulatory actions were necessary to fulfill the Commission's obligations under the Atomic Energy Act[39] for the RTR sabotage event. The Commission found that the 10 CFR Part 100 reactor siting dose criteria were appropriate as an upper dose limit for the potential radiological consequences resulting from sabotage events. The worst case sabotage event for the test reactor resulted in doses that were a fraction of the 10 CFR Part 100 reactor siting dose criteria. The extent of facility damage assumed in the worst case sabotage scenario may bound the facility damage of beyond-design-basis external events. Considering

[39] The Commission is directed by Section 104c of the Atomic Energy Act to impose only such minimum amount of regulation of the licensee as the Commission finds will permit the Commission to fulfill its obligations under the Act to promote the common defense and security and to protect the health and safety of the public and will permit the conduct of widespread and diverse research and development.

the bounding nature of the worst case sabotage scenario, would require additional assessment. Given that assessment, it may be possible to conclude that the radiological consequences resulting from any severe external event will likely not exceed the test reactor licensing criteria.

As discussed above, for the test reactor, NRC staff is performing additional assessments of the resiliency of the facility's emergency power and active decay heat removal systems to a severe seismic and or other external event. If the additional assessment demonstrates that the emergency power and active decay heat removal systems remain capable to perform their functions following such events, NRC staff is unlikely to pursue additional actions. If the additional assessment concludes otherwise, that is, emergency power and active decay heat removal would not be available following a severe seismic or external event, then NRC staff will use the appropriate regulatory process to determine whether to implement additional features to prevent or mitigate these severe seismic or external events at the test reactor.

The additional assessments for MURR, MITR, and NBSR will be performed by NRC staff with expertise in the areas of seismology, hydrology, structural analysis, and RTR design and operation. These assessments are expected to be limited in nature, will use existing budgeted resources, and will be prioritized considering other high-priority work that is being performed by these staff. Upon completion, NRC staff will provide a Commissioners' Assistants Note presenting the results of the assessments and, if appropriate, a discussion of the work necessary to resolve any remaining concerns at these facilities.

Public Comments Received During March 13, 2015, Public Meeting to Discuss Staff's Preliminary Assessment of Applicability of Fukushima Lessons Learned to Facilities Other than Operating Power Reactors

This enclosure provides a list of public comments received during the March 13, 2015, public meeting. The public meeting was broken into six major sections. In the first five sessions, the staff described its preliminary assessment of the Fukushima lessons learned for the following facilities:

- Session 1 consisted of the NRC staff's overview of its preliminary assessment for spent fuel storage installations, transportation packages, and decommissioned reactors and complex materials facilities.
- Session 2 consisted of the NRC staff's overview of its preliminary assessment for fuel cycle facilities.
- Session 3 consisted of the NRC staff's overview of its preliminary assessment for radioactive materials users and irradiators.
- Session 4 consisted of the NRC staff's overview of its preliminary assessment for low-level waste disposal facilities, uranium recovery facilities, and uranium mill tailings.
- Session 5 consisted of the NRC staff's overview of its preliminary assessment for research and test reactors.

After each of the first five sessions the staff provided stakeholders the opportunity to ask questions and provide comment on the staff's preliminary assessment. For session six the staff did not make any presentations and stakeholders were provided an opportunity to ask questions and provide comments on the topics discussed in the staff's preliminary assessment.

The following table provides a description of the comments received, the staff's response to the comment, and, if applicable, the changes made to the staff's assessment based on the comment. Each comment was provided a unique identifier based on the session in which it was received. For example the first comment received in session 1 is provided the identification number S1-1, and the sixth comment in session 5 is provided the identification number S5-6. The unique identifier can also be found in the March 13, 2015, meeting summary (Agencywide Documents Access and Management System (ADAMS) at Accession No. ML15077A410).

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
S1-1	Donna Gilmore - Recommends the staff qualify the conclusion for the independent spent fuel storage installation (ISFSI) such that it is clear that it does not apply to the ISFSI	The staff has processes in place to address aging management in the renewal period and is actively involved in research activities related to identifying aging issues associated with spent fuel storage systems. The staff's Fukushima assessment is based on casks meeting regulatory requirements and if the casks meet the regulatory requirements the staff has confidence that its

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	renewal period. Commenter referenced an August 5, 2014, meeting summary associated with potential cracking in dry shielded canisters (DSCs) over time and a concrete workshop that suggests degradation of concrete over time is possible.	conclusion that no radiation-related death or injuries or significant offsite contamination is expected. As a result of the comment NRC staff added a discussion to the spent fuel storage and transportation section that addresses the requirements for spent fuel storage system license and certificate renewal and the assessment's applicability to the renewal period.
S1-2	Donna Gilmore – Consequences of a DSC crack are not addressed. Inspections have not been developed to find such cracking. Paper should address both issues.	The staff is continuing its research to ensure aging mechanisms are addressed. In the initial license term, there are fabrication and loading inspections and processes in place to ensure integrity of the storage system at the time it is put into service. In recognition that storage systems will need to be in service into their renewal period, the NRC and industry are involved in research to ensure potential aging mechanisms, including stress corrosion cracking, are identified and that appropriate detection methods are developed. Staff anticipates that non-destructive inspection techniques will be developed within approximately five years. Staff finds these activities are sufficient to ensure that the ISFSIs continue to meet NRC regulations in the renewal period. The staff is also developing a report to evaluate the consequences of fuel aging which will be issued in the near term. NRC staff did not identify changes to the assessment as a result of this comment.
S1-3	Marvin Lewis – Paper does not recognize that Fukushima happened and that transportation accidents happen.	The purpose of the staff's assessment is to address lessons learned from the Fukushima accident. The staff's transportation assessment, as discussed in the draft white paper (see ADAMS Accession No. ML15042A367), is based on NUREG-2125, "Spent Fuel Transportation Risk Assessment," which recognizes transportation accidents happen. NRC staff did not identify changes to the assessment as a result of this comment.
S1-4	Marvin Lewis – NUREG 2125 uses optimistic assumptions.	The staff believes that the assumptions in NUREG 2125 are conservative. NRC staff did not identify changes to the assessment as a result of this comment.
S1-5	Marvin Lewis – Meeting does not have adequate public participation.	The staff issued a press release on March 2, 2015, to inform interested stakeholders about the meeting and provided a draft of the paper on February

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
		23, 2015. 35 non-NRC personnel either registered for the webinar or participated via phone in the meeting. NRC staff did not identify changes to the assessment as a result of this comment.
S2-1	Bob Link – Paper states that generic letter (GL) for fuel facilities will be issued in March 2015, and during the presentation the staff stated that GL will be issued in April.	Paper changed to reflect expectations regarding issuance of GL in 2015.
S3-1	George Rudy – Paper does not mention initiators associated with terrorist attacks or aircraft impacts.	The Paper's focus is for Fukushima lessons learned. Terrorist attacks and aircraft impacts are addressed in separate analysis performed by the staff in response to the September 11, 2001, terrorist attack. For example, the staff has done extensive analysis for theft and diversion and has implemented additional security requirements. The staff is continuously assessing the security threat and the current regulatory for security to determine whether additional regulatory actions are needed. NRC staff did not identify changes to the assessment as a result of this comment.
S4-1	Donna Gilmore – Does the earthquake map provided on slide 56 of the staff's presentation (ADAMS Accession No. ML15070A530) take into account the new long-term forecast for California from the US Geological Survey issued on March 10, 2015? If the paper does not, should the assessment be changed to reflect the latest seismic information?	No, the paper does not take into account the March 10, 2015, USGS earthquake forecast for California. The staff's assessment is based on a qualitative analysis of its regulations and the staff's conclusions are based on the licensing processes that ensure that facilities other than operating power reactors are robust and that systems structures and components that are necessary to protect the public will function properly in the event of an earthquake. The staff continuously assesses new information through its operating assessment programs to ensure that new information like the USGS earthquake forecast for California are evaluated to determine whether or not the NRC has to take action to ensure public safety. NRC staff did not identify changes to the assessment as a result of this comment.
S4-2	Donna Gilmore – Paper does not address aging management and	The staff's assessment is based on the staff's practice that NRC's regulatory processes are periodically updated to address new information. In addition, for

Comment Number	**Person Providing Comment and Description of Comment**	**Response to Comment**
	recommendation made to add assumptions used in the paper.	renewed licenses, the staff looks at the effects of aging mechanisms to ensure that they are properly addressed before a license is renewed. The staff's assumption is therefore based on licensees' compliance with the current regulations and that this compliance will be maintained into the future. For low level waste (LLW) facilities, the NRC regulations address long-term protection of the public by ensuring barriers are in place to ensure the public is protected. As a result of this and other comments NRC staff added a discussion in the SECY paper to discuss aging management of dry cask systems.
S4-3	George Rudy – Should the Department of Energy's Waste Isolation Pilot Project be addressed in the paper?	The Waste Isolation Pilot Project is not regulated by the NRC and is therefore outside the scope of the paper. NRC staff did not identify changes to the assessment as a result of this comment.
S5-1	George Rudy – Did the research and test reactor assessment assume post-accident monitoring systems?	No, the assessment did not consider the effects of Fukushima lessons learned on the research and test reactors (RTRs) post-accident monitoring system. RTRs are not required to have a post-accident monitoring system like operating power reactors. They are required to have, as part of their design, equipment and instrumentation necessary for the safe shutdown of the facility and needed to maintain the facility safely shutdown. What specific equipment and instrumentation is required at a RTR is determined during the licensing process for each facility. This includes nuclear instrumentation systems and level and temperature indication to assess the condition of the core, and area radiation monitoring systems to monitor radiation levels in the vicinity of the reactor. The higher powered reactors have more facility equipment and instrumentation including backup and emergency power supplies and decay heat removal systems. The staff did assess Fukushima lessons learned for RTRs and, as stated in the paper, concluded that additional assessment is needed to be performed for the 3 RTRs that are greater than 2 megawatts in power. NRC staff did not identify changes to the assessment as a result of this comment.
S5-2	Marvin Lewis – Did the scope of the	The scope of the paper is limited to NRC-licensed RTRs. The Atomic Energy

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	paper include military and Department of Energy reactors?	Act of 1954, as amended, does not give the licensing and oversight authority to the NRC for the DOE and DOD reactor facilities. Therefore, they are not normally licensed by the NRC. In special cases, the NRC has agreed to provide licensing and oversight of a specific DOD facility following the establishment of an appropriate legal agreement. For example, licensing and oversight of the Armed Forces Radiobiological Research Institute is the responsibility of the NRC and it was included in the assessment. NRC staff did not identify changes to the assessment as a result of this comment.
S5-3	Ruth Thomas – Does the staff consider Defense Nuclear Safety Board regulated facilities and DOE facilities?	The scope of the paper is limited to NRC – licensed RTRs. See the response to Comment S5-2
S5-4	Ruth Thomas – The paper and presentation uses the term "very unlikely." Is the term very unlikely defined in probabilistic terms?	The use of the terms in the paper such as "highly unlikely" is a qualitative term and is not based on a quantified probabilistic risk assessment (PRA) for RTRs. The staff used highly conservative assumptions as documented in the paper. For example, for RTRs less than 2 megawatts the staff assumed a complete loss of coolant and determined that air cooling alone was sufficient to remove decay heat and prevent fuel damage such that fission products are released. NRC staff did not identify changes to the assessment as a result of this comment.
S5-5	George Rudy – Do RTRs have a no fly zone around them?	No, RTRs do not have a no-fly zone around them. Although outside the scope of the meeting the staff has done extensive review of NRC regulated facilities based on lessons learned from the September 11, 2001 terrorist attack. Based on the location of the RTR the staff performs additional reviews. For example one of the RTRs is located next to a runway. The licensee analyzed an aircraft impact event as one of the design base external events and provided additional information regarding hazards associated with the event. NRC staff did not identify changes to the assessment as a result of this comment.
S5-6	Ray Lutz – Requested that audio file be	Audio files from the March 13, 2025, meeting were made publicly available on

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	made available.	the NRC website. The files can be found at the following link: http://www.nrc.gov/reactors/operating/ops-experience/japan/japan-meeting-briefing.html
S5-7	Ray Lutz – RTRs should be reviewed for effects to the public from terrorist attacks.	The scope of the staff's assessment is limited to Fukushima activities. In Enclosure 1 to this document, Section 8, Subsection VII, the staff provides a discussion of the 2006 RTR security assessments and their interrelationship with the Fukushima lessons learned assessment. In addition to the 2006 security assessment, the impact to the public has also been considered in RTR sabotage studies and aircraft impact assessments. NRC staff did not identify changes to the assessment as a result of this comment.
S6-1	George Rudy – Does the NRC coordinate Fukushima lessons learned with other Federal agencies so that other Federal agencies can take advantage of NRC's work for reviewing their regulated facilities? For example, is the NRC coordinating with DOE regulated facilities?	NRC shares Fukushima lessons learned and other operating lessons learned with other Federal agencies and with the international community so that the agencies can take advantage of NRC's work and the NRC can take advantage of their work. NRC staff did not identify changes to the assessment as a result of this comment.
S6-2	Ace Hoffman – NRC staff is exhibiting gratuitous ignorance as it applies to aircraft impacts and dry cask storage breaches and is not properly assessing these events.	The staff has performed detailed security assessments of aircraft impacts on spent fuel storage systems. The staff has discussed the results of these assessments with the appropriate licensee or vendor and did not identify any changes to the ISFSI design as a result of these evaluations. NRC staff did not identify changes to the assessment as a result of this comment.
S6-3	Ace Hoffman - There is an absence of discussion of the need for prompt evacuation or for the impacts to the Price Anderson insurance protection in the event of an accident.	The staff's assessment does include discussions of whether or not changes to emergency plans are needed as a result of applying Fukushima lessons learned to facilities other than power reactors. The staff did not identify changes needing to be made in this area as a result of its review. The staff did not identify changes needing to be made to Price Anderson

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
		protections based on the result of its assessment.
S6-4	Donna Gilmore – The Fukushima dry cask storage system is different than U.S. dry cask storage systems. Did the NRC consider what would happen to U.S. cask designs had they been at Fukushima?	The staff evaluated the existing US NRC approved dry cask storage system designs, grouped by common design characteristics, as part of its assessment of applying Fukushima lessons learned to these facilities. The type of spent fuel storage cask in use at Fukushima would be most similar to the casks evaluated with common characteristics of vertical orientation and bolted closure. All spent fuel storage cask designs must meet the regulatory requirements of 10 CFR Part 72. NRC staff did not identify changes to the assessment as a result of this comment.
S6-5	Donna Gilmore – Humboldt Bay does not have air vents to remove heat because of low heat load. The San Onofre and Callaway underground design have air vents which would let water into the system. The staff should consider this.	As documented on page 11 of the staff's assessment, the staff did consider spent fuel storage systems in the following conditions: 1) fully submerged in water, 2) partial flooding, and 3) blockage of the vents of the canister overpack. The results of the steady-state thermal analysis for a vertical cask system and a horizontal cask system indicate that a release from the confinement is not expected. NRC staff did not identify changes to the assessment as a result of this comment.
S6-6	Donna Gilmore – What is the remediation plan if a DSC fails?	NRC regulations are developed to ensure that containment is maintained in the spent fuel storage system. The DSC is one component of a canister-based spent fuel storage system. The staff has performed safety assessments and determined that the probability of a DSC failing and the associated consequences are very low. NRC staff did not identify changes to the assessment as a result of this comment.
S6-7	Donna Gilmore – The security assessments referenced in response to S6-6 should be provided to the public	NRC staff continually balances the public's need to know with making the same security assessment information available to potential terrorists. The staff has provided classified briefings on the security assessments to individuals with a need to know. There is no plan to make the security assessments publicly available at this time.

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
		NRC staff did not identify changes to the assessment as a result of this comment
S6-8	Ray Lutz – Presentation does not discuss Fukushima lessons learned.	The staff is seeking comments not on the presentation but on the draft white paper that was provided as part of the meeting notice. The presentations were meant to be a brief summary of the staff's assessment and do not include the analysis found in the draft white paper.
S6-9	Ray Lutz – Slide 12 from the meeting handouts states that Near Term Task Force Fukushima lessons learned number 2 regarding the need to reevaluate and upgrade design basis flood and seismic design basis events is not applicable. This does not appear to be accurate.	The staff did evaluate flooding and seismic scenarios as part of its assessment as documented in the draft white paper and found the consequences to be low. Based on this comment the staff changed the recommendation for ISFSI applicability for Fukushima item 2 from "Not applicable" to "No action."
S6-10	Ace Hoffman – Aircraft hazards and terrorist events should be considered by the staff.	The scope of the paper is limited to Fukushima lessons learned. However, as noted in response to comments S3-1, S5-7 and S6-2, the NRC has performed security assessments for some of the facilities discussed in its assessment.
S6-11	George Rudy – A naturally generated missile or seismic event could cause a degraded DSC to fail, therefore, the staff needs a preplanned mitigation plan.	The scope of this assessment was to evaluate the applicability of the lessons learned from the event at Fukushima to NRC-regulated facilities or stakeholders, other than commercial power reactors. Earthquakes and tornado missile events were assessed. Based on the staff's assessment the staff did not identify needed changes to its current regulatory approach for ISFSIs and therefore did not identify a need for a mitigation plan.
S6-12	Marvin Lewis – Do the spent fuel storage cask designs consider the enrichment of Uranium 235?	Yes. The staff reviews and captures in technical specifications requirement for critical fuel characteristics which include the enrichment, the time the fuel has spent in the reactor, and the condition of the fuel itself. NRC staff did not identify changes to the assessment as a result of this comment.
S6-13	Marvin Lewis – 5/8 inch thick DSC are	Spent fuel storage cask systems must meet the requirements in 10 CFR Part

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	deficient.	72 to be authorized for use by the NRC. The DSC is only one component of the spent fuel storage system. The NRC has approved storage systems with DSC designs that are approximately ½-inch thick. The DSCs are transported and stored in overpacks that protect the DSC from environmental effects such as high winds and tornado borne missiles. In addition the cask system is evaluated against a non-mechanistic tip over. The cask system is designed to demonstrate structural adequacy for this event. The staff found that this tip over assessment bounds the consequences of an earthquake. The staff concluded that the current regulatory requirements ensure the safe and secure storage and transportation of spent nuclear fuel.
S6-14	Donna Gilmore – Commenter expressed concern that San Onofre is in the process of changing the spent fuel pool cooling system to a spent fuel pool island such that ocean cooling will not be needed.	Specific changes to the spent fuel pool cooling system are outside the scope of the draft white paper. The draft white paper documents the staff's assessment of decommissioned reactors' spent fuel pools to determine if changes are needed based on Fukushima lessons learned. The draft white paper concludes changes to the NRC's oversight of these facilities are not necessary at this time. Note: Subsequent to the meeting the Office of Nuclear Reactor Regulation's San Onofre project manager followed up with the commenter to provide additional information in response to the commenter's concern about changes to the San Onofre spent fuel pool cooling system
S6-15	Donna Gilmore – In a previous discussion NRC indicated that reactor vessels that experienced inner wall cracking did not leak. Commenter is not concerned about water leaking from the DSC but rather gas leaking through DSC cracks.	See response to comment S1-2
S6-16	Donna Gilmore – Concerned that environmental monitors are being removed from around ISFSIs. Decommissioned reactors and ISFSI should have real time environmental	Based on the staff's review of Fukushima lessons learned the staff did not identify a need for real time environmental monitors for ISFSIs or decommissioned reactors. Limits for the release of radioactive material from ISFSIs can be found in 10 CFR 72.104, "Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS." Emergency planning

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	monitors.	requirements for ISFSIs can be found in 10 CFR 72.32, "Emergency plan."
S6-17	Donna Gilmore – San Onofre ISFSIs are only 100 ft above the sea level. With global warming this is not enough elevation.	The NRC continuously reviews operating experience information to determine if changes to its oversight of licensees need to be made. As described on page 26 of the draft white paper, for more than 30 years, the NRC has implemented improvements in its emergency preparedness and incident response programs especially upon reviewing lessons learned after severe natural disasters. The draft paper goes onto describe actions taken during and in the aftermath of Hurricane Katrina. The staff also notes that for the San Onofre ISFSIs to be impacted by a sea level rise of 100 ft, the effects of global warming would have to be orders of magnitude higher than currently predicted.
S6-18	Donna Gilmore – Commenter concerned that NRC is not reviewing the San Onofre ISFSI before it is installed and will only perform inspections prior to its operation.	Implementation of the ISFSI at San Onofre has gone and will undergo thorough NRC review. NRC authorizes storage of spent nuclear fuel at an ISFSI under two licensing options: site-specific license and general license. At San Onofre, the licensee plans to use the general license provisions. Under the general license provision, the licensee intends to use an NRC-approved cask. An NRC approved cask is one that has undergone a technical review of its safety aspects and has been found to be adequate to store spent fuel within the bounding site characteristics evaluated in the safety analysis report. The licensees using such approved casks must show that the cask conditions and technical specifications can be met and document the evaluation as required by 10 CFR 72.212, "Conditions of general license issued under § 72.210." The licensee's evaluation is subject to NRC inspection.
S6-19	Donna Gilmore – NRC processes should be changed to require NRC approval prior to installing an ISFSI or making major changes to the spent fuel pool.	The requirements for installing an ISFSI can be found at 10 CFR Part 72 and evaluation requirements before a licensee can make a change to a spent fuel pool can be found at 10 CFR 50.59. These requirements ensure that appropriate NRC review is performed before an ISFSI is installed or if a change to a spent fuel pool reaches a threshold for which prior NRC approval is required.
S6-20	George Rudy – There is inadequate	See response to comment S5-1.

Comment Number	**Person Providing Comment and Description of Comment**	**Response to Comment**
	post-accident consideration for research and test reactors.	
S6-21	Ray Lutz – The conclusion for the spent fuel storage section of the paper states that the no further regulatory action or study is necessary. This conclusion does not recognize that the staff needs to continuously assess its regulations in light of Fukushima lessons learned.	The conclusion in the spent fuel storage section of the draft white paper reflects that no further regulatory action or study is necessary as a result of this assessment of the lessons learned from the Fukushima event. The NRC continuously evaluates if its regulations and guidance needs change or update in light of new pertinent information and will continue to do so.
S6-22	Ray Lutz – ISFSIs should have a regulatory requirement to have Emergency Planning Zones.	ISFSIs are required to have in effect an emergency plan that meets the requirements of 10 CFR 72.32 (for a specific licensed ISFSI) or 10 CFR 50.47 (general licensed ISFSI). The NRC believes that emergency planning requirements in 10 CFR 72.32 are adequate for ISFSIs. The *Federal Register* notice for 10 CFR Part 72 (60 FR 32430, 32435) describes the NRC's rationale for not including Emergency Planning Zones for ISFSIs. The staff did not identify changes needing to be made in this area as a result of its review.
S6-23	Ray Lutz – Assessment is deficient and needs to be redone because it does not provide enough of a discussion describing the Fukushima cask designs and compare them against the US designs to identify lessons learned.	See response to comment S6-4.
S6-24	Donna Gilmore – NRC should require a license amendment for ISFSI changes or for changes to the spent fuel pool.	See response to comment S6-19

Discussion of Email Comments Received Associated with the Staff's Preliminary Assessment on the Applicability of Fukushima Lessons Learned to Facilities Other than Operating Power Reactors

This enclosure provides a list of public comments received via email associated with the NRC staff's preliminary assessment of the applicability of Fukushima lessons learned to facilities other than operating power reactors. The 14 sets of email comments received as a result of the March 13, 2015, meeting can be found in an enclosure to the meeting summary (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15077A410).

The following table provides a synopsis of the email comment documented in the March 13, 2015, meeting summary, NRC staff's response to the comment, and any changes made to the NRC staff's assessment based on the comment. Each comment was provided a unique identifier based on the origin of the comment. For example, the Missouri at Columbia Research Reactor (MURR) comments are listed as the first set of email comments documented in the March 13, 2015, meeting summary. MURR's first comment is provided comment number E1-1. The "E" designation is used to indicate that the basis for the comment is an email. This designation was used as an aid to distinguish the comments received verbally during the meeting and those provided via email.

Comment Number	**Person Providing Comment and Description of Comment**	**Response to Comment**
E1-1	Missouri at Columbia Research Reactor (MURR) - On page 69, the 3rd bullet under Electrical Power, "3-out-of-31 RTRs may require ac power to replenish inventory because of seismic activity or boil-off." Assume by replenishing inventory you mean water inventory for the reactor and/or spent fuel pools? As it currently stands, MURR does not need electrical power to fill the pool in an emergency situation on a lowering of pool level. MURR has a dedicated water line with a manual isolation valve that will provide virtually an unlimited quantity of water to the	Given that this comment is related to the immediate assessment performed by NRC staff following the Fukushima accident and was limited to the determination adequacy of a facility's design basis assessment, the commenter's statement is correct under the assumed conditions. As a result of the comment NRC staff changed the referenced text to: "2-out-of-31 RTRs may require ac power to replenish the reactor coolant inventory lost as a result of seismically induced failure of reactor coolant system or boil-off."

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	pool.	
E1-2	MURR - On page 71, the 5th bullet under Spent Fuel Pool Cooling, “Typically, spent irradiated and excess fuel is stored dry in the reactor pools, or in dedicated spent fuel pools provided by design.” Not sure what is meant by “Stored dry in the reactor pools”? How can you store something dry in a water pool?	Comment is correct. Comma added to correct sentence structure.
E1-3	MURR - Page 71, 3rd bullet under Combustible Gas Control, “The higher powered RTRs have dedicated hydrogen control system that are operated during reactor operation; however, upon reactor shutdown, those systems are no longer required to prevent the formation of combustible or explosive concentrations of hydrogen.” MURR does not require a hydrogen control system. MURR attached response to a relicensing request for additional information regarding this subject and it has been accepted by the NRC.	Comment is correct and the report has been modified appropriately.
E2-1	National Institute of Standards and Technology – NIST provided several pages of clarifying comments regarding the capabilities of the test reactor.	The staff incorporated the comments as appropriate when the comment provided clarification or corrected language regarding the capability of the test reactor.
E3-1	Massachusetts Institute of Technology	The staff recognizes the concern presented by the commenter. In a

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	(MIT) – The paper discusses 10 CFR Part 20 limits in the context of a maximum hypothetical accident (MHA). The part 20 limits were not specifically written to be applied to accident scenarios.	separate activity associated with updating licensing requirements for research reactors under the rulemaking process, NRC staff is considering developing accident dose criteria for research reactors.
E3-2	MIT – The paper places radiological consequences in context of a facility's effective power. The concern is that a facility's effective power could be seen as precedence for using values other than license limits for other regulatory activities.	The NRC staff recognizes the comment's concern regarding the use of values other than license limits for regulatory determinations. However, the topic of this paper is lessons learned from the accident at Fukushima Daiichi, which evaluates accident conditions caused by external events of greater magnitude than those considered sufficiently credible for use in licensing. Therefore, the NRC staff's assessment considered parameters such as effective power to ensure the assessment reflected realistic effects. No changes were made as result of this comment.
E4-1	PBorchmann - Comment discusses Superflex and industry preferences for cheaper and less extensive site upgrades. The comment also references two non-concurrences associated with the draft assessment.	The staff believes the majority of the comment refers to a different white paper that was issued in the process of developing COMSECY-14-0037, "Integration of Mitigating Strategies for Beyond-Design Basis External Events and the Reevaluation of Flooding Hazards." The COMSECY is available in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML14238A616. Two non-concurrences associated with COMSECY can be found in the enclosures to the COMSECY. The staff's draft assessment on the applicability of Fukushima lessons learned to facilities other than power reactors did not have non-concurrences.
E4-2	PBorchmann – Comment mentions that stakeholders in southern California are concerned about Section 7 of the white paper regarding decommissioned	Comment does not provide specific concerns regarding the decommissioning portion of the white paper, therefore, no change to the document was identified by the staff.

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	reactors.	
E5-1	Alliance for a Green Economy, Citizens' Environmental Coalition, Nuclear Information and Resource Service – Citing the West Valley Nuclear site in Western NY, challenges statements regarding timeliness in decommissioning activities as related to the West Valley Demonstration Project (WDVP).	NRC staff recognizes that the WVDP is a unique case and different from other NRC licensed complex sites. In fact, the WVDP Act of 1980 (i.e.; the Act) gives the NRC an advisory and a monitoring role with respect to the WVDP activities. Under the Act, the Department of Energy (DOE) shall decontaminate and decommission: (a) facilities of the Western New York Service Center in West Valley in which high-level waste was solidified or stored, (b) the facilities used in the solidification of waste, and (c) any material and hardware in connection with project. As a result of the Act, the NRC's WVDP license was put into abeyance. When DOE's work under the Act is completed, the NRC license will be reactivated and, New York State Energy Research and Development Authority will be responsible for the license. Therefore, due to the complex nature of the site, the interdependence of multiple Federal and State authorities, as well as the unique nature and process of decommissioning activities and associated costs, more time is needed to complete decommissioning of all WVDP facilities and land. As a result of comments received regarding the WVDP, NRC staff added a discussion of this project to the assessment found in Enclosure 1
E5-2	Commenters found in comment 5-1, stated that the NRC should list the decommissioned facilities that the white paper is evaluating and mention West Valley, Savannah River, Los Alamos, and Hanford Sites. The commenters request whether or not NRC staff consider these sites low risk.	The decommissioned facilities evaluated in this draft white paper included all decommissioned facilities with active NRC or Agreement State licenses. These facilities can be found in the 2014 Status of the Decommissioning Annual Report, an NRC document, which can be accessed in ADAMS at Accession No. ML1429A239 (http://pbadupws.nrc.gov/docs/ML1429/ML14294A239.pdf). DOE has the lead (direct responsibility) for West Valley until the NRC license for that site is reactivated—the NRC license is currently in abeyance. The other sites referenced are also DOE's direct responsibility.

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
		The NRC's role at the Savannah River, Hanford and West Valley sites involves overview and consultation on implementation issues. These sites are therefore not appropriate for consideration in an NRC paper assessing whether further regulatory action is necessary to address lessons learned from the Fukushima Daiichi accident at NRC licensed facilities.
E5-3	Commenters found in comment 5-1, stated that a vulnerability assessment for earthquakes and climate change for complex decommissioning sites should be performed. The commenters referred to WVDP as indicated in the statements: "The reality is that at West Valley and many other complex material facilities, there are interred materials that are anything but securely contained;" and … "Radioactive releases from West Valley nuclear waste site could also occur with extreme rainfall events, and conditions which result from climate change will inevitably increase the risk and consequences at the site."	NRC staff assessed potential risk at complex material sites licensed by the NRC or Agreement States. As stated in the draft white paper, the potential risk from natural phenomena is low under severe scenario assumptions using Fukushima lessons learned potential dose impacts. The potential risk is not expected to exceed Environmental Protection Agency (EPA) Protective Action Guideline levels for members of the public under an emergency situation. Regarding, the West Valley Demonstration Project (WVDP), the natural phenomena, including containment and external events (e.g.; severe rainfall and flooding), were adequately addressed in DOE's Final Environmental Impact Statement (EIS) (which was issued in January 2010 and can be found in ADAMS at Accession No. ML100750122) and earlier Safety Analysis Report (SAR) (which can be found in ADAMS at Accession No. ML032940574). For example, in Chapter 4 of the EIS and Chapters 2 and 9 of the SAR, DOE evaluated potential public hazards and events related to containment and loss of confinement, fire hazard, criticality, seismology, meteorology, and surface/subsurface hydrology.
E5-4	Commenters found in comment 5-1, citing West Valley, challenge statements in the white paper that a decommissioning plan is required to be developed.	Due to the unique and complex nature of the WVDP decommissioning facility, the decommissioning plan is being submitted in in two phases. NRC staff reviewed the Phase I Decommissioning Plan (issued on February 25, 2010, ADAMS Accession No. ML10036000) which is being implemented by DOE. The Phase II decommissioning Plan is anticipated by 2020.
E5-5	Commenters found in comment 5-1, indicate that the white paper contains	The EPA Protective Action Guideline levels under emergency situations were used as guidelines to evaluate necessary regulatory actions. In

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	no data, facts, or science supporting the discussion that external events pose low risk to decommissioned facilities.	addition, NRC staff evaluated "Decommissioning Plans" or "License Termination Plans" describing all safety aspects to protect workers, the public, and the environment from potential release of radioactivity. The NRC licensees assess site location, climate conditions, and potential safety issues related to events such as extreme rain or flooding conditions, tornadoes, and high wind events. Since the location of radioactive materials during decommissioning are either on building surfaces, in soil, or could be stored in piles or in containers for ultimate packaging and disposition, the risk from potential radioactivity releases for licensed facilities was found to be low. Section 5 of the draft white paper also referred to studies already carried out and reported as NUREGs or technical reports.
E5-6	Commenters found in comment 5-1, request the West Valley site-specific emergency preparedness plan.	The DOE is responsible for operating and implementing Phase I decommissioning activities. The DOE is self-regulating and has an emergency plan in place.
E5-7	Commenter found in comment 5-1, question whether a site evaluation was ever performed for the West Valley Nuclear facility.	In the Final WVDP EIS, DOE addresses the site evaluation and status
E5-8	Commenters found in comment 5-1, challenge whether the West Valley Nuclear facility considered external events and that NRC would be able to mitigate the release of radioactive material from the site.	DOE is responsible for mitigation and cleanup. The NRC's current role in Phase I of the decommissioning activities associated with WVDP is advising DOE and monitoring their decommissioning and decontamination activities. The Phase II decommissioning plan is anticipated to be developed around 2020. Therefore, defining the specific role of each authority/agency in Phase II is premature.
E6-1	Donna Gilmore – Near term task force (NTTF) recommendation 1 not adequately addressed because the draft white paper does not address aging management of dry cask storage.	See response to comments S1-1 and S1-2.

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
E6-2	Donna Gilmore – The dry cask storage at Fukushima are superior to those licensed in the US.	See response to comment S6-4.
E6-3	Donna Gilmore – NTTF recommendation 2 is not adequately addressed. Aging management, new seismic science, and climate change not adequately assessed for ISFSIs.	NRC staff disagrees with this comment. See response to comment S6-9 for additional detail.
E6-4	Donna Gilmore – NTTF recommendation 8 regarding emergency planning not adequately addressed.	NRC staff disagrees with this comment. See response to comment S6-22 for additional detail.
E6-5	Donna Gilmore – NTTF recommendation 11 regarding emergency planning not adequately addressed. Continuous remote radiation monitoring systems should be required for ISFSIs.	The paper's focus is for Fukushima lessons learned, not the performance of individual licensees. The staff does routinely assess licensee performance and did not identify changes needing to be made in this area as a result of its review.
E6-6	Donna Gilmore – Provided a comment related to NTTF recommendation 12 regarding strengthening the oversight process. There is a lack of defense in depth for ISFSIs and NRC oversight for utilities that have shown a pattern of mismanagement should be increased.	See responses to E6-1 through E6-5 above.
E7-1	Joe Holtzman – Supports the comments from Donna Gilmore and indicates that he totally disagrees with the staff conclusions in the draft white paper.	NRC staff disagrees with the comment. For responses to the endorsed comments see responses to comment sets E6 and S6.
E8-1	Marv Lewis – General comments that	NRC staff disagrees with the comment.

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	NRC has not adequately addressed Fukushima lessons learned and provides general concerns including transportation of other hazardous materials on roads and railways along with radioactive shipments.	As the comments relate to the transport of other hazardous materials on road and railways, the Department of Transportation has specific limitations related to shipment routing and mixing of different hazardous materials on the same conveyance in the Hazardous Materials Regulations, 49 CFR Parts 100 to 177. The responsibility of the safe transportation of radioactive materials is shared between NRC and the Department of Transportation through a memorandum of understanding (44 FR 38690).
E9-1	Ruth Thomas – general comments such as draft white paper: 1) does not define very low risk, 2) does not address the consequences of an airplane hitting a reactor, and 3) does not address the vulnerability of non-power reactors.	Regarding the definition of risk – the assessment found in Enclosure 1 is generally qualitative in nature and therefore, risk is not generally described in terms of an overall number. NRC staff's assessment does reference NUREG 2125, "Spent Fuel Transportation Risk Assessment," which does provide a quantitative analysis. Where qualitative terms are used NRC staff attempts to provide perspective. For example, NRC staff defines the magnitude of the consequences of a natural phenomena hazard and external events as "low" for the NRC's spent fuel and transportation assessment if no radiation deaths or injuries are expected and no offsite contamination is expected. Regarding airplane hitting a non-power reactor, and vulnerability of non-power reactors, several other comments were received in this area. See response to questions S3-1, S5-7, and S6-2.
E10-1	Paul Frey – Provided power point presentation on economic damage to California due to meltdown and fuel pool fires at Diablo Canyon.	The paper is limited to Fukushima lessons learned for facilities other than power reactors. NRC staff notes that regarding spent fuel pools, several assessments have been made including those referenced in Section 7 of Enclosure 1. These assessments include NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," and more recently, SECY-13-0112, Enclosure 1, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor" (ADAMS Accession

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
		No. ML13256A342).
E11-1	Bruce Campbell – Disagrees with overall conclusions of the draft white paper.	NRC staff acknowledges the comment.
E11-2	Bruce Campbell – Does not believe that ISFSIs are properly addressed, and asserts that NRC staff needs to ensure best quality casks like those in Germany. NRC staff also needs to assure reputable transport company and a place to store the waste.	Regarding the ISFSI design portion of the question see response to comment S6-13. Regarding the portion of the comment associated with storing high-level waste, the long-term plan for the disposal of spent fuel is within the purview of the President and Congress; the current plan established by Congress is codified in the Nuclear Waste Policy Act (NWPA). While the NRC recognizes that the implementation of the current plan faces political and societal challenges, it is not the NRC's role to set policy in this area. The NRC has developed regulations for the independent storage of spent nuclear fuel in accordance with the NWPA. Staff believes that the evaluation adequately addresses the recommendations from the NTTF and did not identify changes needing to be made in this area as a result of its review. Regarding the portion of the comment regarding transportation, the responsibility of the safe transportation of radioactive materials is shared between NRC and the Department of Transportation through a memorandum of understanding (44 FR 38690). These regulations ensure safe transportation of radioactive materal.
E11-3	Bruce Campbell – Low level waste site leak tritium and other radioactive material and are not properly regulated.	Tritium release may occur at a LLW Disposal facility but the risk from such release would be insignificant and may represent a small fraction of the permissible public dose limit. Nevertheless, if such release occurs, the concerned State and operator should take appropriate action (e.g., deploying geo-membranes) to limit surface-water infiltration.

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		The current assessment pertains to natural phenomena and natural events by assuming that radioactive materials may be released due to severe events as such: flooding, surface erosion, or earthquake. The main conclusion derived in this analysis is that, under these severe conditions, the risk is low and the potential dose impact to members of the public would not exceed EPA Protective Action Guidelines limits for the public under emergency situation. Considering the low dose conversion, the dose from a tritium leak would be far below the 10 CFR Part 61 safety requirements for public dose limit.
E11-4	Bruce Campbell – Mill tailings are not adequately regulated.	The regulations that the NRC has established for mill tailings are found in 10 CFR Part 40, Appendix A, "Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the Extraction or Concentration of Source Material from Ores Processed Primarily for Their Source Material Content." In addition, 10 CFR Part 20 dose criteria for protection of the public apply to mill tailings.
E11-5	Bruce Campbell – Nonpower reactors not adequately regulated. Lawrence Livermore Lab plutonium experiment released radioactivity to the environment.	The scope of the paper is limited to NRC-licensed RTRs. The Atomic Energy Act of 1954, as amended, does not give the licensing and oversight authority to the NRC for the DOE and DOD reactor facilities. Therefore, they are not normally licensed by the NRC. In special cases, the NRC has agreed to provide oversight of a specific DOD facility following the establishment of an appropriate legal agreement. Furthermore, the NRC has no knowledge of any reactors located at the LLNL.
E11-6	Bruce Campbell – Fuel cycle generic letter is deficient. There should be specific site analysis to ensure safety.	The generic letter requests fuel cycle facility licensees to provide information on site specific analyses of natural phenomena hazards and demonstrate how they meet the NRC regulations and requirements of their license. Compliance with the regulations provides reasonable assurance of public health and safety.
E11-7	Bruce Campell – NTTF recommendation associated with upgrading design basis components	See response to comment S6-9.

Comment Number	Person Providing Comment and Description of Comment	Response to Comment
	related to seismicity and flooding should not be considered "not applicable."	
E12-1	Marni Magda – It is inappropriate to allow San Onofre to have buried canisters left on a bluff for 20 years	See response to comment S6-18.
E13-1	Michel Lee – Document uses overly optimistic assumptions. Climate change will exert a multiplier effect on aging mechanisms such that risks will increase.	For comments related to aging management of spent fuel storage systems, see response to comments S1-1 and S1-2. For LLW disposal NRC staff performs assessments of facilities. In performing the assessments, NRC staff assumes that LLW packages/containers would degrade over time and, as a result, radioactive materials may reach the receptor through multiple pathways. NRC staff considers natural phenomena events such as flooding due to climate change for LLW facilities. Aging mechanisms that may lead to release of radioactive material from LLW packages/containers are also considered in NRC staff's assessment.
E13-2	Michel Lee – Economic risk placed on taxpayers is not addressed.	The purpose of the draft white paper is to address potential impact of natural phenomena on regulations of other than operating power reactors facilities. In other words, it is intended to address necessary regulatory actions to ensure protection of the health and safety of the public.
E13-3	Michel Lee – Paper does not address the radioactive release of entire fuel cycle.	The purpose of the draft white paper is to address potential impact of natural phenomena on regulations of other than operating power reactors facilities. Therefore, NRC staff did not address the radioactive release of the entire fuel cycle.
E13-4	Michel Lee – Paper is based on flawed assumptions including assuming nothing will go wrong, compliance with regulations ensures safety, and corporations will address radioactive releases from their facility.	The paper conservatively assumes that extreme natural phenomena hazards happen and based on this performs an assessment of whether or not additional regulatory action is needed. As noted in the paper NRC staff identified two areas where additional assessment is needed: 1) generic letter for fuel cycle facilities, and 2) additional assessments for high-power reactors. NRC staff does assume that licensees will meet their regulatory

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		responsibilities; otherwise, appropriate regulatory action will be taken.
E13-5	Michel Lee – Paper does not address cyber security	Cyber security is outside the scope of the Fukushima lessons learned assessment. NRC staff is taking several actions in this area for both operating power reactors and other facilities the NRC regulates.
E14-1	PBorchmann supplemental comment – Draft white paper does not recognize climate change impacts.	Discussion of climate change found in response to comments S6-17, E5-3, E5-5, E6-3 and E13-1. The analysis performed by the staff considered severe flooding (independent of the cause of the flood) including the following: • fully and partially submerged ISFSI cask (see Enclosure 1), • lessons learned based on Hurricane Katrina (see Enclosure 1)