Note: This paper was updated on March 2, 2015 to include the Staff's Assessment of NRC-Licensed Non-power Reactors (changes made to pages 3, and pages 62 through 86 added)

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

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Summary

This draft white paper is being provided as opportunity to comment on the staff's assessment of the applicability of Fukushima lessons learned to facilities other than operating power reactors that are regulated by the U.S. Nuclear Regulatory Commission (NRC, or Commission). This draft white paper has been developed based on Commission direction to consider the applicability of lessons learned from the event to "non-operating reactor and nonreactor facilities." The facilities and NRC licensees included in this paper include:

- Spent Fuel Storage and Transportation Systems
- Fuel Facilities
- Radioactive material uses,
- Irradiators licensed in accordance with Title 10 of the *Code of Federal Regulations* (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 research and test reactors

The white paper is being provided to support public outreach and its contents should not be interpreted as official agency positions. The paper has not been subject to management and legal reviews and approvals. The NRC staff expects that this topic will be addressed in a future memorandum from the staff to the Commission.

Introduction

Shortly following the March 11, 2011, earthquake and tsunami that initiated the accident at the Fukushima Dai-ichi nuclear power reactor facility in Japan, the NRC staff was directed (Tasking Memorandum – COMGBJ-11-0002 – NRC Actions Following the Events in Japan," dated March 23, 2011[1]) to undertake a number of near-term and longer term actions. The most visible activity was the creation of a Near-Term Task Force (NTTF) to perform a review of the events and the possible implications for the safety of United States nuclear power plants. On July 12, 2011, the NTTF issued its report, entitled, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan." The NRC prioritized the NTTF recommendations and other activities related to lessons learned from the Fukushima accident for nuclear power plants in the United States, issued orders and requests for information to NRC licensees for nuclear power reactors, and has provided periodic updates on the status of those activities.

The tasking memorandum issued shortly after the Fukushima accident, also included as one of the longer term activities that the staff should assess the applicability of the lessons learned to non-operating reactors and non-reactor facilities. The staff performed limited assessments to ensure no immediate safety concerns were identified and took some measures, such as performing an independent verification of fuel facility licensees' ability to prevent and/or mitigate

[1] The March 23, 2011, memorandum is available in the Agencywide Document Access and Management System (ADAMS) under Accession No. ML110820875

the consequences of events which could challenge the safety or licensing bases of those facilities. With insights having been gained from the NRC activities related to power reactors and the results from the inspection of fuel cycle facilities, the NRC staff more fully evaluated issues and possible actions related to other NRC licensed materials, devices and facilities. This draft white paper documents the staff's preliminary evaluations of lessons learned from the Fukushima accident for certain classes of NRC licensees and Agreement State licensees.

Discussion

The series of events at Fukushima Dai-ichi nuclear power plant resulting in core melt was initiated by the earthquake and subsequent tsunami. The flooding inundated the emergency power systems, exceeding the design bases for the facility. The event response revealed a number of circumstances which had been unanticipated primarily because they were considered to be extremely unlikely, and thus not planned for.

The Commission established the NTTF in response to the event and chartered the group with conducting a systematic and methodical review of NRC processes and regulations and determining if the agency should make improvements to its regulatory system. It was 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. ML11231A723) the 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) the NRC Staff developed long term plans to take action on the recommendations. As stated in SECY-11-0137, the Staff will continue to evaluate the applicability of lessons learned to licensed facilities other than power reactors (e.g., research and test reactors, 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.

With insights having been gained from the NRC activities related to power reactors such as the seismic and flooding walkdowns and the results from the inspection of fuel cycle facilities, the NRC staff more fully evaluated issues and possible actions related to other NRC licensed facilities, devices and materials. The types of licensee events that were assessed under this guidance 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 fire impacts to determine if existing regulatory requirements appropriately address such hazards. In addition to the evaluation of initiating events and external hazards, the various NRC licensed facilities, devices, and materials were qualitatively assessed in terms of other Fukushima-related policy issues and the findings and recommendations in the NTTF Report, and other studies and evaluations performed domestically and internationally. The assessments looked at the broad scope of the lessons learned from Fukushima and were not necessarily focused on specifics, which tend to be discussed in the context of nuclear power reactors.

As part of the assessment the NRC staff involved in the review generally performed a desk review and tabletop discussion to screen and evaluate specific types of NRC-licensed facilities, devices and materials. These evaluations were performed through document review of license requirements and regulatory framework, and internal meetings with subject licensee experts from the program offices, supplemented if needed, by staff from regional offices, Office of Nuclear Security and Incident Response, and Office of Nuclear Regulatory Research. The level of effort for these evaluations depended on the nature of the licensed facility, device or material (e.g., amount of material, complexity, etc.). The staff's assessments that follow are for the following general areas:

1. Spent Fuel Storage and Transportation
2. Fuel Facilities
3. Radioactive Material Uses
4. Part 36 Irradiators
5. Low-Level Waste Disposal Facilities
6. Uranium Recovery Facilities and Uranium Mill Tailings
7. Decommissioning Reactors and Complex Materials Facilities
8. Non power research and test reactors – updated on 3/2/15

For the eight classes of licensees discussed in this draft white paper, the staff's overall conclusion is that the existing regulatory processes are sufficient to ensure protection of public health and safety. The staff's evaluation identified the following two areas where work will continue to ensure that Fukushima lessons learned are appropriately considered:

- The 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. The staff expects to issue a generic letter on this subject in March 2015

- For the three non-power research and test reactors that have thermal power ratings in excess of 2 Megawatts (MWs), the staff is performing additional assessments to determine if it is necessary to include additional features to prevent or mitigate certain beyond design basis seismic and external events (updated on 3/2/15).

If responses to the fuel facilities generic letter or the additional assessments for the research and test reactors that exceed 2 Megawatt Thermal (MWt) determine that additional features are appropriate to prevent or mitigate external events, then the staff will use the appropriate regulatory process to implement these features.

For the other regulated facilities discussed in this paper that staff has determined that no additional analysis or regulatory action is needed based on its review of the lessons learned from the Fukushima Dai-ichi accident.

1. Spent Fuel Storage and Transportation Systems

I. Current Regulatory Framework

The main hazard at spent fuel storage facilities and systems and of transportation packages is radiation exposure. These facilities, systems, and packages have multiple components and use specific materials of construction to protect the workers, public, and the environment. NRC's responsibility is to review, approve, and inspect the design of storage facilities, systems, and transportation packages to ensure that doses are as low as reasonably achievable. NRC also ensures that spent nuclear fuel at storage facilities and in transit are secured. The 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 (10 CFR Part 72)[2] and transportation package designs (10 CFR Part 71)[3] allows for issuing certificates of compliance, general licenses, or specific licenses. Certificates of compliance are granted to vendors that own 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>

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. For the design of the dry cask storage systems, vendors consider natural phenomena hazards 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 can have a general or specific license to store spent nuclear fuel (i.e., an independent spent fuel storage installation (ISFSI)). For a general license, the utility is responsible for verifying that the design parameters of the selected cask system bound the natural phenomena hazards that may occur at their site. For any site-specific parameter(s) that are not bounded by the vendor's design, the utilities must apply for a

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[2] 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."
[3] 10 CFR Part 71, "Packaging and transportation of radioactive material."

specific license and verify the adequacy of the cask design to withstand the specified natural phenomena hazards for a given site.

There are currently 55 general licensees and 15 specific licensees of independent spent fuel storage installations in the United States. The two types of independent spent fuel storage installations are dry storage[4] or wet storage[5] (i.e., spent nuclear fuel pool). There is only one wet storage facility in the United States (i.e., G.E. Morris) and its fuel is more than 20 years old.[6]

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 the DOT[7] and 10 CFR 71.5, "Transportation of Licensed Material." The DOT is the competent authority for all transportation matters in the United States. The NRC's responsibility is to ensure that packages for fissile materials and Type B quantities[8] 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 also supports the DOT on the review of packages that would be used in the United States by 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.[9]

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 international transportation regulations[12] are revised.

Inspection and Oversight

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

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[4] A vertical, horizontal, or underground storage system placed on a concrete pad or vault.
[5] Concrete structure with a liner.
[6] NRC Inspection Report 072-00001/11-01, dated 3/13/2011.
[7] 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.
[8] A Type B quantity is defined in 10 CFR 71.4, "Definitions."
[9] TS-R-1, "Regulations for the Safe Transport of Radioactive Material, 2009 Edition Safety Requirements."

In 2010, as a result of an audit from the Office of Inspector General (OIG),[10] the staff started implementing recommendations for improving the spent nuclear fuel storage and transportation program. The improvements included developing a qualification journal for inspectors of independent spent fuel storage installations and revising 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." A qualification journal was issued and the Inspection Manual Chapter 2690 was revised to better define the frequency for performing inspections.[11]

Emergency Preparedness

a. Storage of Spent Nuclear Fuel

Contents of an emergency plan for spent fuel storage facilities are included in 10 CFR 72.32. Per 10 CFR 72.32, an application for a specific-license of an independent spent fuel storage installation must include an emergency plan 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 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.

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 emergency plan required by 10 CFR 50.47 shall be deemed to satisfy" the 10 CFR 72.32 requirements for an emergency plan. There are no current regulatory requirements for establishing emergency planning zones in the emergency plan 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,[12] including severe weather conditions as well as 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.

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[10] OIG-11-A-12, "Audit of NRC's Oversight of Independent Spent Fuel Storage Installations Safety," dated May 19, 2011. (ADAMS Accession No. ML11140A132)

[11] G201208/EDATS: OEDO-2012-0171, Memorandum from Stephen D. Dingbaum RE: "Update on Status of Implementation of Recommendations Contained in Audit of U.S. Nuclear Regulatory Commission's Oversight of Independent Spent Fuel Storage Installations Safety (OIG-11-A-12)," dated October 12, 2012. (ADAMS Accession No. ML12279A222)

[12] 10 CFR Part 73, "Physical protection of plants and materials," Appendix D, "Physical protection of irradiated reactor fuel in transit, training program subject schedule."

Security

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

a. Storage of Spent Nuclear Fuel

Both, specifically and generally licensed independent spent fuel storage installations have to 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 ISFSI 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 terrorists' attacks of September 11, 2001, the NRC issued security orders to all spent nuclear fuel storage facilities to impose additional 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,
2. review and approval of shipment routes of spent nuclear fuel, [13]
3. requirements for personnel accessing the shipment,[14] and
4. specific requirements for transportation of category 1 and category 2 quantities of radioactive material. [15]

II. Post–Fukushima Event Evaluations/Assessment

The staff performed evaluations on both transportation systems and storage designs to determine the impact natural phenomena hazards. For the transportation assessment, the staff used NUREG-2125, "Spent Fuel Transportation Risk Assessment,"[16] published in January 2014, as the basis for the qualitative evaluation.

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[13] 10 CRF 73.37, "Requirements for physical protection of irradiated reactor fuel in transit."
[14] 10 CFR Part 73.38, "Personnel access authorization requirements for irradiated reactor fuel in transit."
[15] 10 CFR Part 37, "Physical protection of category 1 and category 2 quantities of radioactive material." NUREG-0561, Revision 2, "Physical Protection of Shipments of Irradiated Reactor Fuel," includes staff's guidance for reviews related to 10 CFR Part 37.
[16] ADAMS Accession No. ML14031A323.

Additionally, the staff participated in a series of table top exercises to evaluate the impact of selected natural phenomena hazards to existing storage designs. The staff did not evaluate site specific designs, since the main design characteristics[17] of storage systems were the same for general and specific licensees. Staff's expertise included knowledge of 10 CFR Part 72 requirements to evaluate the adequacy of the storage design in the following areas:

1. confinement evaluation,
2. criticality safety evaluation,
3. materials evaluation,
4. shielding evaluation,
5. structural evaluation, and
6. thermal evaluation.

The staff divided its qualitative assessment into the following phases:

Phase 1.	**Determine initial applicability of Near Term Task Force (NTTF) recommendations to spent fuel storage and transportation certificate holders and licensees.**
Phase 2.	Determine commonalities among the design characteristics of the storage systems. (Rely on NUREG-2125 as basis for the transportation qualitative assessment.)
Phase 3.	Determine qualitative magnitude of consequences of natural phenomena hazards and external events.
Phase 4.	Document findings, determine path forward, and develop input.

The qualitative assessment is based on a single canister and assumes that all spent fuel systems are in compliance with 10 CFR Part 72 regulations. Loading activities (including on-site transportation), as well as the independent spent fuel storage installation reinforced concrete supporting pad were not evaluated as part of this assessment. However, loading activities, external hazards, and potential consequences are evaluated as part of the NRC licensing and oversight process. Similarly, the adequacy of the independent spent fuel storage installation reinforced concrete supporting pad is evaluated during the licensing review process.

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,' provided 12 recommendations. The staff considered these recommendations for applicability to independent spent fuel storage installations and the transport of spent nuclear fuel and evaluated to determine if any future action is warranted. The following table summarizes the result of the review.

[17] i.e., wet storage or dry storage, horizontal or vertical system orientation, and welded or bolted closure.

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
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).	Not Applicable
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
4	The Task Force recommends that the NRC strengthen station blackout mitigation (SBO) capability at all operating and new reactors for design-basis and beyond-design-basis external events.	No Action
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
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
7	The Task Force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	No Action
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].	Not Applicable
9	The Task Force recommends that the NRC require that facility emergency plans address prolonged SBO and multiunit events.	No Action
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
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
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

b. Phase 2. Determine commonalities among the design characteristics of the storage systems.

Based on the current regulatory framework for licensing, ISFSIs were categorized into two major groups, general licenses and specific licenses.[18] The staff further grouped the casks systems into different design characteristics in order to simplify the qualitative assessment. Design characteristics included:

1. Orientation on the ISFIS pads (i.e., vertical, horizontal, or underground);
2. Closure type (i.e., bolted or welded);[19] and
3. Vents on overpack (i.e., vented or not vented).

After grouping the storage systems, the staff performed the qualitative evaluation of the impacts of selected natural phenomena and external events.

c. Phase 3. Determine qualitative magnitude of consequences of natural phenomena hazards and external events.

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

The staff's initial assessment 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 consequence events. The 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. Based on the

[18] 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 ISFSI concrete pad.
[19] Vertical systems only.

qualitative evaluation, the staff determined that these events would also result on low radiological consequences. The following summarizes the staff's qualitative evaluation:

1. **Seismic** - Per 10 CFR Part 72.102[20] and 10 CFR Part 72.103[21] 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 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.[22]

2. **Flood** - The staff considered scenarios such as a 1) canister fully submerged in water, 2) partial flooding, and 3) blockage of the vents of the canister overpack.[23] The worst case scenario in which the heat transfer would not be adequate is a partial flood of a storage system and this scenario would not cause a radioactive release to the environment. For the third scenario, the staff assumed that all vents of an overpack system were blocked with debris, the debris were 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. After an event, the licensee should ensure that cask systems remain in the certified condition.

3. **High Winds** – For this scenario, the 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. The majority 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.[24]

   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 miles per hour (mph) (Texas) to 290 mph (Southeastern Atlantic Coastline) depending on the location of the independent

[20] 72.102, "Geological and seismological characteristics for applications before October 16, 2003, and applications for other than dry cask modes of storage."
[21] 72.103, "Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003."
[22] Independent spent fuel storage installation pad.
[23] Storage canisters are not allowed to have events. Systems that include and overpack, the overpack may have vents for cooling purposes.
[24] ASCE/SEI 7-05, "Minimum Design Loads for Buildings and Other Structures."

spent fuel storage installations. The dry cask storage systems would usually be less than 33 feet high. Missiles generated by hurricanes were not considered in this portion of the assessment, since the tornado assessment would bound high winds generated-missile events.

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, provides guidance that the 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.

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, the 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 (see Section II.4 of this document).

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.[25] 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,[26] after the event at Fukushima. NRC inspectors reviewed scenarios, such as station blackout, seismic, tornado, flood, and fire. The staff found that the fuel would not melt as a result of these events due to its limited heat load.

III. International Experiences

Since the March 11, 2011, seismic and tsunami events at 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, IAEA, and others. 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 utilized, as applicable, for facilities within the United States.

IV. Conclusions and Recommendations

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

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

The defense-in-depth philosophy is embedded in 10 CFR Part 71 and 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.[27] Therefore, the staff concludes that no further regulatory action or study is necessary.

2. Fuel Facilities

I. Overview of Fuel Cycle Facilities

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 isotope uranium-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 shown in the following table:

Code of Federal Regulations	**Subject**
10 CFR Part 20	Radiation Protection
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 uranium hexafluoride (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-

[27] 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."

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.

Hazards: As with mining and milling, the primary risks associated with conversion are chemical and radiological. 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 (HF) gas (chemical hazard), a non-radioactive, extremely toxic substance.

Plants: 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 or U-238 (U_{238}), 0.7 percent uranium-235 or U-235 (U_{235}), and < 0.01 percent uranium-234 or U-234 (U_{234}). The fuel for nuclear reactors has to have a higher concentration of U_{235} than exists in natural uranium ore. This is because U_{235} is "fissionable," meaning that it starts a nuclear reaction and keeps it going. 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 additional shielding. Another hazard 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 (Scheduled shutdown)
- 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). The NRC was directed by Congress 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. Most at risk from these hazards are the plant workers.

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 & Wilson Nuclear Operations in Lynchburg, VA
- Mixed Oxide Fuel Fabrication
  - Shaw AREVA MOX Services, LLC in Aiken, SC (Construction permit issued, construction ongoing)

D. Uranium Hexafluoride Deconversion

Process: As uranium-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 and fluoride compounds (such as hydrogen fluoride) are hazardous at low levels of exposure.

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

E. Academic and Other Institutions

Academic and other institutions use nuclear material in classroom demonstrations, laboratory experiments, research, and to provide health physics support to other institutional nuclear materials users. 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. These facilities are licensed in accordance with 10 CFR 30, 40, or 70 depending on the type of materials possessed.

Hazards: These licensees have limited amounts of materials and have demonstrated by Agency 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 due to the limited amount of materials possessed and the primarily sealed nature of the material, the impact to the public and the environment is limited. Furthermore, each of these licensees have obligations under Occupational Safety and Health Administration, but these are separate from their regulatory commitments under 10 CFR Part 70.

II. Post–Fukushima Event Evaluations/Assessment

The staff performed a systematic evaluation and inspection of selected fuel cycle facilities to confirm that licensees were in compliance with regulatory requirements and license conditions; and to evaluate their readiness under natural phenomena hazards (NPH) events and other licensing bases events related to NPH. The staff's assessment considered the NTTF recommendations to determine whether additional 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, the 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. The results of the evaluation allow the staff to conclude 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, for the Honeywell Metropolis Work Facility, the inspections identified potentially significant safety issues and the staff took immediate steps to ensure corrective actions were implemented. In addition, the staff identified 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. The staff is in the process of developing a generic letter 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.

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

On March 31, 2011, the 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, the 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, the staff reviewed licensing documents, including the safety assessments and emergency plans. 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 and/or mitigate emergencies during selected beyond licensing basis natural events and extended loss utilities that support onsite systems. In the review of licensing basis events, the NRC considered the following NPH: seismic, flooding, and high winds (caused by hurricanes or tornadoes). The NRC also evaluated onsite fires because seismic events may cause facility fires as a result of failures of plant equipment. Particular attention was given to earthquakes and flooding because of recent events and significant advancements in the state of knowledge of these hazards.

Following 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, the NRC staff identified 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 were in compliance with their licensing basis and regulatory requirements. The NRC inspectors opened unresolved items to further assess whether the evaluated licensees are in compliance with license conditions, the requirements of 10 CFR 70.61 and 10 CFR 70.62(c), regarding NPH events accident sequences. A summary of the results of the inspections can be found in Table 1. The staff has determined that for all the facilities inspected, except the Honeywell Metropolis Works Facility, due to consideration of inherent seismic capacity in buildings, 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 Work Facility, the NRC determined that the site Emergency Response Plan underestimated the amount of uranium hexafluoride and hydrogen fluoride that could potentially be released during credible seismic events or tornadoes. Specifically, the inspectors identified 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 which 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).

Table 1 Summary of TI 2600/015 results

	Facility	Summary of Issues Identified
Part 76	Paducah (ADAMS Accession No. ML12131A437)	• Tornados were not considered a credible event due to 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 as a result of a tornado event are bounded by others safety basis events.
Part 70	AREVA (ADAMS Accession No. ML12122A094)	• URI 70-1257/2012-006-01 was opened to further evaluate whether the licensee is in compliance with the requirements of 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 is in compliance with the requirements of 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 is in compliance with the requirements of 10 CFR 70.62(c) and the performance requirements of 10 CFR 70.61 regarding accident sequences that are a result of natural phenomena events.
	Nuclear Fuel Services (ADAMS Accession No. ML12122A186)	• URI 70-143/2012-06-01 was opened to further evaluate whether the license is in compliance with Table 2.2 of the license application regarding management measures for IROFS PREP-A and PREP-B. • URI 70-143/2012-006-03 was opened to further evaluate whether the licensee is in compliance with the requirements of 70.62(c) and 70.61 performance requirements regarding natural phenomena events accident sequences.

	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 to 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.
Part 40	Honeywell (ADAMS Accession No. ML12222A163)	• URI 40-3392/2012-006-01 was opened to evaluate whether the MTW 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. • 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 provide 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>

The NTTF report, 'Recommendations for Enhancing Reactor Safety in the 21st Century,' provided 12 recommendations. These recommendations have been considered 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.

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
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 (Refer to Generic Letter discussion)
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

4	The Task Force recommends that the NRC strengthen station blackout mitigation (SBO) capability at all operating and new reactors for design-basis and beyond-design-basis external events.	Not Applicable
5	The Task Force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable
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
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.	Not Applicable
9	The Task Force recommends that the NRC require that facility emergency plans address prolonged SBO and multiunit events.	Not Applicable
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
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
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

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

As a result of the inspections, the staff is issuing a generic letter due to the generic applicability of the URI's 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.

The staff maintains continuous engagement with industry and other stakeholders through public meetings and presentations during the Fuel Cycle Information Exchange. The staff plans to issue the generic letter before spring 2015 and expects to close out the review of the responses early 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. An administrative review of the readiness of GTCM facilities was performed (ADAMS Accession No. ML14111A087). The recommendations of the NTTF were reviewed for applicability, as well as considerations identified during the review of Fuel Cycle Facilities. On the basis of this review, the continued operation of GTCM licensees does 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.

III. International Experience

The 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, the staff participated in an International Atomic Energy Agency (IAEA) consultancy and technical meeting to develop an IAEA 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, the staff in coordination with the Office of International Programs, held a meeting with the Director and Principal Inspector of the Decommissioning, Fuel, and Waste Program (Office of Nuclear Regulation) from United Kingdom to broaden cooperation with the United States in areas related to the fuel cycle, storage, disposal, decommissioning, and activities associated to the post-Fukushima event for fuel cycle facilities.

IV. Conclusions and Recommendations

As a result of the systematic and methodical evaluation of fuel facilities and greater than critical mass licensees in light of the lessons learned from Fukushima, the staff concludes that the current regulatory approach and requirements for fuel cycle licensees provides reasonable assurance of adequate protection of public health and safety. The 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. The staff expects to issue a generic letter on this subject in March 2015.

3. Radioactive Material Uses

I. Current Regulatory Framework

The NRC regulates approximately 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 approximately 18,900 licensees. The quantities possessed by medical and academic licensees 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. Sealed sources are required to satisfy rigorous design and performance criteria. In order 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. Devices used in these settings are designed to return to a safe or shielded mode in the event of a loss of power, air pressure, or other external factors. The sources and devices are designed to survive normal conditions of use and likely accidental conditions.

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. Self-shielded irradiators incorporate many engineering features to protect individuals from unnecessary radiation exposure. These devices are usually designed for use in a laboratory environment, i.e., inside a building, protected from the weather, and without wide variations in temperature and humidity.

Licensees also 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 industrial irradiators is now described in 10 CFR Part 37, “Physical Protection of Category 1 and 2 Quantities of Radioactive Materials”. This regulation establishes security requirements for the use and transport of the most risk-significant quantities of radioactive materials.

Licensing

NRC regulations applicable to devices containing byproduct material and to NRC radiation safety evaluations, are found in 10 CFR Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders"; 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations"; 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 2 Quantities of Radioactive Materials"; 10 CFR Part 40, "Domestic Licensing of Source Material"; 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions"; 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"; 10 CFR Part 170, "Fees for Facilities, Materials, Import and Export Licenses, and Other Regulatory Services Under the Atomic Energy Act of 1954, as Amended"; and 10 CFR Part 171, "Annual Fees for Reactor Licenses and Fuel Cycle Licenses and Materials Licenses, Including Holders of Certificates of Compliance, Registrations, and Quality Assurance Program Approvals and Government Agencies Licensed by the NRC."

The NRC's NUREG-1556 technical report series, Consolidated Guidance about Materials Licenses, provides 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:

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, and
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. Where applicable, additional guidance on source security and safety culture have been included.

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. The NRC or Agreement State will perform a safety evaluation 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 (SSD) Registration Certificate. Sealed sources are required to satisfy rigorous design and performance criteria. In order 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.

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 additional 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[28] 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.

Inspection and Oversight

For the fabrication and use of sealed sources and devices used by radioactive material users, the NRC assures compliance with the regulatory requirements through both periodically scheduled and special inspections. The 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 provide 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, it is necessary to develop written procedures to

[28] 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.

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 rads) to the thyroid; or (2) develop an emergency plan for responding to a release of radioactive material which includes 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. At present, most medical and academic licensees do not possess quantities of radioactive materials that exceed the values described in § 30.72, Schedule C.

Licensees who utilize radioactive materials for portable and fixed gauges, industrial radiography, and well logging are required to develop and follow operating and emergency procedures. Each year, a number of gauges experience equipment failures because of corrosion caused by the harsh environmental conditions. Following 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 considers theft or the loss of a gauge, radiography, or well logging source to be a situation which could constitute a substantial hazard. 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.

Sealed sources are required to be leak tested at intervals not to exceed 6 months and if found to be leaking in excess of 185 Bq (5 nanocuries) of removal contamination, the device must be immediately withdrawn from use and placed in a safe storage location until it is decontaminated, repaired, or disposed of according to NRC requirements. Special authorization may be granted by NRC to applicants to decontaminate a facility contaminated by a leaking sealed source.

II. Evaluations and Assessments Prior to Fukushima

While most equipment that use sealed sources are very rugged, accidents occur which may 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 fires 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; most, but not all, were subsequently recovered by divers. In other instances, moisture density gauges have been run over by a truck or tractor trailer with minor damage to the storage container. In those few instances where the source was dislodged from the source holder, the source was quickly located and re-inserted into the source holder and protective shielding.

In 2006, a nuclear pharmacy reported flood damage due to 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 material was not affected by the tornado. No other instances involving category 1 or 2 medical or academic sources to external events are documented in the Nuclear Material Event Database. There are no instances in the Nuclear Material Event Database documenting the loss of licensee control of category 1 industrial, medical or academic sources due to severe weather, earthquake, or flooding. While no category 2 medical or academic sources have been lost, there are rare instances of category 2 industrial source losses.

Safety evaluations for sources and devices used in industrial settings are generally evaluated during the initial license review. The evaluation of the sealed sources and devices include a review of the design, manufacturing, prototype testing, and proposed uses as discussed under licensing. Devices used in these settings are designed to return to a safe or shielded mode in the event of a loss of power, air pressure, or other external factors. The sources and devices are designed to survive normal conditions of use and likely accidental conditions. However, devices occasionally are damaged or break during normal operation and the operator must initiate an emergency response. In the vast majority of these instances, the source is safely returned to its shielded position and appropriate event notifications are made to regulatory authorities.

For more than 30 years, the NRC has implemented improvements 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 located in those states potentially impacted 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, Department of Energy (DOE), 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 locate any missing or misplaced radiation source. The 2005 Hurricane Season Lessons Learned Task Force proposed as 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.

III. Post-Fukushima Event Evaluations/Assessments

While 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 there is radioactive material in any of this floating debris.

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 to 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 National Source Tracking System (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 conduct inspections and investigations, communicate information to other government agencies, and verify legitimate ownership and use of nationally tracked sources. Since the Fukushima accident, the NRC also has developed a mapping tool using Google Earth to provide 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. Currently, 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 the NSTS and Google Earth are regularly used to monitor and verify source security after natural events like earthquakes, wildfires, tornados, hurricanes and flooding.

Staff evaluations of radioactive material licensees were performed 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 provides 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:

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	The greatest concern is the loss of licensee control of radioactive material. During the license application process, the location of all

External Event	Outcome	Assessment
Seismic	Challenge to structures in which unsealed and sealed sources are used or stored. Gamma knife equipment must be checked by the manufacturer before patient use.	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 National Source Tracking System in accordance with 10 CFR 20.2207. The theft or the loss of a gauge could constitute a substantial hazard and therefore be reportable under 10 CFR 20.2201. Emergency plans that address natural phenomena, 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.
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 due to 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	Challenge to transport of sealed and unsealed source material.	

The NTTF report, ‘Recommendations for Enhancing Reactor Safety in the 21st Century,’ provided 12 recommendations. These recommendations have been considered 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.

Table 2. 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

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 structures, systems and components (SSCs).	Not Action
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
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-basis external events.	No Action
5	The Task Force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable
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
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.	Not Applicable
9	The Task Force recommends that the NRC require that facility emergency plans address prolonged SBO and multiunit events.	No Action
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
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
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

IV. 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 provide 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 upwards 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[29] of cobalt-60 with activities that range from 9 PBq (250 KCi) to 1 EBq (30 MCi).

There are several types of irradiators licensed by the NRC and Agreement States. 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. There are over 50 wet-source-storage irradiators in use in the United States 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 which 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

[29] 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).

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 Materials.” This regulation establishes security requirements for the safe use and transport of the most risk-significant quantities of radioactive materials.

Licensing

NRC regulations applicable to devices containing byproduct material and to NRC radiation safety evaluations, are found in 10 CFR Part 2, "Rules of Practice for Domestic Licensing Proceedings"; 10 CFR Part 19, "Notices, Instructions, and Reports to Workers; Inspections and Investigations"; 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 2 Quantities of Radioactive Materials”; 10 CFR Part 71, "Packaging and Transportation of Radioactive Material"; 10 CFR Part 170, "Fees for Facilities, Materials, Import and Export Licenses, and Other Regulatory Services Under the Atomic Energy Act of 1954, as Amended"; and 10 CFR Part 171, "Annual Fees for Reactor Operating Licenses, and Fuel Cycle Licenses and Materials Licenses, Including Holders of Certificates of Compliance, Registrations, and Quality Assurance Program Approvals and Government Agencies Licensed by NRC."

The NRC’s NUREG-1556 technical report series, Consolidated Guidance about Materials Licenses, provides 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 changes in regulatory policies and practices. The draft revision should be finalized in 2015.

Facility design and available equipment must provide 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 should 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 geographical location of the facility. 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 provide 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 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, especially during an earthquake. 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 whether panoramic wet-source-storage gamma irradiators are constructed, equipped and operated in accordance with the license application.

Emergency Preparedness

The licensee is required to draft and follow 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, a tornado, flooding, or other phenomena as appropriate for the geographical location of the facility.

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 ten seconds, the sources must automatically return to their shielded position. One facility is equipped with backup electrical generators with 5-7 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 licensees emergency or abnormal operating procedures, NRC notification is required when sources are stuck in an unshielded position; any fire or explosion in the radiation room; damage to the source racks; structural damage to the pool liner or wall; and abnormal water loss or leakage 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. 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 United States 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. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 36 was first published in 1993 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. The majority of 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 immediately notify both local law enforcement and the NRC.

The current assessment reviewed the generic facility design and available equipment to determine if there are sufficient engineering controls and barriers 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.

There are currently fifty-five panoramic, wet-source-storage irradiators and one underwater pool irradiator each containing a minimum of 37 PBq (1 MCi) of cobalt-60 licensed for use in the United States. Nine of these irradiators are located 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 which causes the radiation source holder to automatically become fully shielded 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 nominally 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" 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° to 27°C (65° to 80°F). 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°C (265°F). 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 5-7 days at approximately 65°C (150°F). The pool water evaporation rate will be slow, but it would take a number of 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°C (1,470°F) for one 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 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. A 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.

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 integrity will be retained and no radioactive material will be released nor will there be any off site radiation exposure.

Earthquake generated tsunamis were evaluated by the NRC because of facilities in proximity to the Pacific Ocean. 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, it was determined that 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 has sustained enough damage that the source rack was destroyed. In addition, the wave velocity of a wind generated (i.e., hurricane) storm surge is less than that associated with a tsunami. Loss of licensee control of the sealed sources will not occur due to a tsunami or flooding.

During the license application process, the location of all category 1 and category 2 sources must be reported to and tracked in the National Source Tracking System (NSTS) which is maintained by the NRC. 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 provide situational awareness in the event of a severe weather event.

Staff evaluations of panoramic irradiator licensees were performed 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 could reasonably be expected to result from the event that would be more severe than previously evaluated. The following table provides 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:

Table 1. 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
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 ground	

	motions.	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. - Emergency or abnormal event procedures must address natural phenomena, including an earthquake, a tornado, flooding, or other phenomena, as appropriate for the geographical location of the facility.
High Wind and Missiles	Challenge to external structures in which irradiator sources are used. No damage to shield integrity.	
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.	
Loss of Power	Source automatically returns to shielded position. Malfunction of source rack possible thus requiring mechanical intervention to return sources to shielded position.	

The NTTF report, 'Recommendations for Enhancing Reactor Safety in the 21st Century,' provided 12 recommendations. These recommendations have been considered for applicability to Part 36 irradiators and evaluated to determine if any future action is warranted. The following table summarizes the result of the review.

Table 2. 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
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
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
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-basis external events.	No Action

5	The Task Force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable
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
7	The Task Force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	No Action
8	The Task Force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs.	Not Applicable
9	The Task Force recommends that the NRC require that facility emergency plans address prolonged SBO and multiunit events.	No Action
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
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
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

IV. Conclusion and Recommendations

NRC staff concludes that Part 36 irradiators are appropriately licensed and provide 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 best available information, the potential is negligible for natural phenomena (e.g., tsunamis, earthquakes, hurricanes, or fire) to result in a loss of control of radioactive material from an industrial irradiator that would have an adverse impact on public health and safety or the environment. Program specific guidance for irradiator licensing has been updated and now include additional guidance on the security of radioactive materials and changes in regulatory policies and practices. As such, the 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) is not high-level radioactive 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. The LLW includes items that have become contaminated with radioactive material or have become radioactive through exposure to neutron radiation. This waste 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 very highly radioactive in certain cases such as parts from 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 it has decayed away and can be disposed of as ordinary trash, or until amounts are large enough for shipment to a low-level waste disposal site in containers approved by the Department of Transportation.

LLW disposal occurs at commercially operated low-level waste disposal facilities that must be licensed by either the NRC or Agreement States. The "Low-level Radioactive Waste Policy Amendments Act of 1985" (the Act) gave the states responsibility for the disposal of their low-level radioactive waste. The Act encouraged the states to enter into compacts that would allow them to dispose of radioactive waste at a common disposal facility. Currently, there are four commercial disposal facilities that accept low-level radioactive waste. Those facilities are: (1) EnergySolutions in Barnwell, South Carolina (2) U. S. Ecology in Richland, Washington, (3) EnergySolutions in Clive, Utah, and (4) Waste Control Specialists, LLC in Andrews County, Texas. 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 U.S. Ecology site is licensed by the State of Washington 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 EnergySolutions site in Clive is licensed by the state of Utah to accept Class A waste only. The facility accepts waste from all regions of the United States. Finally, the Waste Control Specialists site near Andrews is licensed by the Texas Commission on Environmental Quality to receive waste in Classes A, B, and C. This site is the most recent of the four, given its opening in 2012, and accepts LLW from Texas, 34 States that do not have operating facilities, and the Federal government.

Storage of LLW requires a 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 as low as reasonably achievable. Actual doses, in most cases, are a small fraction of the NRC limits.

Licensing

The NRC and 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 in handling and using of LLW materials.

Regulations and Guidance Documents

The main NRC regulations for shallow land disposal were promulgated in 10 CFR Part 61. These regulations are currently undergoing revision. The staff issued several guidance documents to assist in the implementation of 10 CFR Part 61 requirements, including the evaluation of disruptive phenomenon (e.g. NUREG-1623). Additionally, in certain cases, guidance for other NRC regulatory programs (e.g., NUREG-1757) may be adapted to the10 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, dated August 16, 1995). Staff 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. 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. It also defined a subset of concentration averaging and encapsulation practices that the NRC staff finds acceptable in determining the concentrations of the radionuclides tabulated in 10 CFR 61.55, "Waste Classification." The next revision of the BTP is expected to be issued in early 2015.

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 authorize the NRC to grant emergency access for LLW disposal at each of the commercial facilities 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 is promulgated 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 Act.

Emergency preparedness considerations for LLW facilities are conducted by the appropriate States and their licensees. 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, tornados, and high speed winds, emergency preparedness should be addressed and potential risks/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 mobilize and 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. Currently, 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 should be periodically updated and use monitoring data to reflect actual site performance while taking additional 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 region or state where the facility is to be located, 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 subpart C of this part. Areas having known natural resources which, if exploited, would result in failure to meet the performance objectives of subpart C of this part 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 provide 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 subpart C of this part 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 subpart C of this part, or may preclude defensible modeling and the prediction of long-term impacts. 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 impact the ability of the site to meet the performance objectives of subpart C of this part 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 impact 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). Currently, 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 impacts during a compliance period of 1000 - 2000 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 10 CFR Part 61 was developed in 1982. The performance objectives and other requirements were designed to limit the impact 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 – limits to radiological concentrations and hazardous characteristics (e.g. flammability, pyrophoricity)
- Siting characteristics – provides 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 – active systems cannot be used for long-term safety

Performance objectives – 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 emplaced in disposal cells and those cells are closed, the waste is covered by approximately 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 low-level waste disposal facility requires a high energy event such as a volcanic event. 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 impact the passive performance of a low-level waste 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 provided for remedial response.

III. Post-Fukushima Event Evaluations/Assessments

In response to the Fukushima Lessons Learned activities, the NRC has performed an assessment of the mitigation of impacts of disruptive events and processes on the disposal of low-level waste. 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 low-level waste disposal facilities. In addition, as a result of the inherent characteristics of the waste and disposal facility designs, low-level waste disposal facilities are inherently resistive to the more likely disruptive processes.

As described earlier, 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 Guidance Documents

The staff has developed a new draft guidance document to support the revisions to 10 CFR Part 61 and to supplement existing guidance. Chapter 2 of the guidance document and its associated appendices provide detailed guidance for determining the proper scope of the analysis (i.e. the FEP process) to incorporate the site-specific analysis of disruptive phenomenon. The 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 impact 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 impact of disruptive processes. Based on the characteristics of low-level waste disposal facilities, the staff expects erosive processes (fluvial and aeolian) to be the most likely of all of the disruptive processes to impact 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 impact of disruptive events on a low-level waste disposal facility. When 10 CFR Part 61 was developed, NRC envisioned that the inadvertent use of the disposal site may occur after closure. In order 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 may 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 (post drilling or home construction). 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). The 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 approximately 0.82 tons/acre-yr. Impacts were calculated for time periods of up to 1,000 or 2,000 years following closure. The maximum estimated dose was approximately 1 mrem/yr.

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 impact from a fire scenario was approximately 32 mrem/yr.

Staff evaluated seismic activities as potential natural disruptive events that may impact current commercial LLW disposal facilities. In this regard, 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, staff concluded that LLW commercial disposal facilities are located in areas with peak ground acceleration (PGA) 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 approximately 15 times, or more, stronger than any anticipated earthquake at United States commercial LLW disposal facilities. Therefore, the potential damage for typical buildings located 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, the staff's predicted seismic hazards events would cause no significant radiological impacts to the public.

External Event	Outcome	Potential Risk L M H	Assessment
Flood	• Challenge to LLW storage and treatment facilities, as well as disposal cells before emplacement of cover. • Challenge to onsite waste	Low	The three LLW disposal sites located 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

	shipments waiting for disposal. • Challenge to waste acceptance criteria for contents of liquids. • May challenge integrity of the disposal cell after emplacement of cover.		particularly from uncovered cells; however; potential hazards/risk would be anticipated low due to 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 located 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 impacts. Further, waste packages and disposal operation at the facility should minimize release of radioactive materials, and therefore, minimizing intrusions and providing additional defense-in-depth for safety. More importantly, the current waste acceptance criteria of waste classes and disposal at the sites should provide additional assurance of low hazard or risk from such external event. External events have been addressed in accordance with 10 CFR Part 61 requirements (see above Sections); therefore, emergency plans 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) additional guidance and site-specific risk/hazard assessments are being developed in the context of 10 CFR Part 61 limited rulemaking. Such site-specific analysis should provide additional protective assurance analysis particularly if climate conditions could change in the long-term future.
Seismic	• Challenge to structures and integrity of containers. • Challenges to vaults and covers to contain LLW and minimize intrusion. • Challenges to integrity of disposal cells due to mechanical stresses or due to water infiltration through faults or cracks. .	Low	
High Wind and Missiles	Challenge to structures in which LLW radioactive material is contained or stored in a transition before disposal.	Low	
Lightning	Challenge to structures that may catch fire during lightning which may contain stored pyrophoric radioactive materials in a transition before disposal.	Low	
Snow and Ice Loads	Challenge to structures in which radioactive material is stored or being treated for disposition.	Low	
Drought	None	Low	
Temperature Extremes	None except for pyrophoric materials stored before disposition.	Low	
External Fire	Challenge to structures in radioactive materials stored in a transition for disposition particularly pyrophoric materials.	Low	
Loss of Power	none	Low	

The NTTF report, 'Recommendations for Enhancing Reactor Safety in the 21st Century,' provided 12 recommendations. These recommendations have been considered for applicability to LLW facilities and evaluated to determine if any future action is warranted. The following table summarizes the result of the review.

Table 2. 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
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).	Not Action
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
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-basis external events.	Not Applicable
5	The Task Force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable
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
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.	Not Applicable
9	The Task Force recommends that the NRC require that facility emergency plans address prolonged SBO and multiunit events.	Not Applicable
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.	Not Applicable
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
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

IV. Conclusion and Recommendations

The combination of historical 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 low-level waste 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 additional 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 processes analysis requirements combined with the site-specific intruder assessment and site stability analyses will ensure that disruptive events are appropriately accounted for in licensing of low-level waste 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. Currently, there is no specific protective action guideline limits required for LLW similar to reactors in order 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 transported 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 currently 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 as U_3O_8, because that chemical compound comprises approximately 85 percent of the yellowcake produced by uranium recovery facilities, and that product 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 additional 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 are typically 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 meager concern because uranium has little penetrating radiation and only moderate or non-penetrating beta (β) and alpha (α) radiation to be considered. Nevertheless, the primary radiological hazard is attributable to the presence of radium in the byproduct material. The fire hazard at uranium recovery facilities may arise from two sources:

(a) Storage, handling, and process use of a combustible liquid solvent in the acid leach process at conventional mills. For example, tri-butyl-phosphate (TBP) when used as a solvent may carry some fire hazards. Other solvents, including hydrocarbon liquids, may also be used. Spills from the extraction vessels and leaks from the solvent storage or the transfer pipelines are not uncommon events. This hazard is typically present at conventional mills as ISR facilities do not normally use solvent extraction.

(b) High-temperature calcination process may also carry some fire hazards. Inadvertent carryover of the combustible material into the calciner and a natural gas leak, where heat is provided by natural gas, are possible causes of fire.

However, safety procedures requirements and training of staff should minimize safety concerns or operational hazards.

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.

Most of the regulations that the NRC has established for this type of byproduct material 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 particular, the criteria in Appendix A cover the siting and design of tailings impoundments, disposal of tailings or wastes, decommissioning of land and structures, groundwater 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.

Mill tailings may 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 may be a large trench or a former mine pit and must meet NRC regulations. These piles are generally 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.

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.

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, the Department of Energy (DOE) was responsible for cleanup at 20 legacy sites. All of these sites are located 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 all surface cleanup and is continuing to clean up groundwater at several sites. The DOE maintains these sites and provides long-term surveillance under a general license from the NRC.

Title II - Sites licensed in 1978 or later

These sites must be licensed by the NRC or an Agreement States. For all uranium sites, the following apply:

- Private companies recover uranium and are responsible for cleaning the UR site contamination;
- Facility design must be submitted for review with the application; and
- Plans to clean up any surface and/or groundwater contamination must be approved by the NRC or the Agreement State in which the site is located.

For conventional mill facilities, the following apply:

- Design of the tailings piles must be submitted for review with the license application;
- Plans to clean up any surface and/or groundwater contamination must be approved by the NRC or the Agreement State in which the site is located;
- 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 groundwater at Title I sites.

There are 11 NRC-licensed uranium mill sites that 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 took over authority for the site as an Agreement State. The Texas Commission for Environmental Quality (TCEQ) licensed a facility at Waste Control Specialists, LLC near Andrews, TX in 2008.

A uranium recovery license is valid for 10 years from the date of the Commission finding, under 10 CFR 40.32. A uranium recovery license can be renewed in 10-year increments. As of April, 2014, the NRC is reviewing 8 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"
- 10 CFR Part 40, "Domestic Licensing of Source Material"
- 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"
- Agreement States' regulations for uranium recovery facilities at Texas, Colorado, and Utah, must be compatible with NRC's above mentioned regulations.

Uranium milling and disposal of the resulting byproduct material by NRC licensees are regulated under Title 10, Part 20, of the Code of Federal Regulations (10 CFR Part 20), "Standards for Protection Against Radiation"; 10 CFR Part 40, "Domestic Licensing of Source Material"; and Appendix A to 10 CFR Part 40. Appendix A 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 conducts inspections of licensed facilities and cover areas such as training of personnel who operate a uranium recovery facility, radiation protection programs, dose records, and ensuring security of nuclear materials onsite and during shipment.

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

- ensure, through inspection, site visits, and review and approval of safety and environmental monitoring reports and data that uranium recovery facility 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 corrective actions to conform with inspection findings or any violation of operational safety procedures; and
- 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 independently monitor radioactive releases on a continuous basis, but performs planned and unplanned visits, and inspections in coordination with the regional staff, 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 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 different scenarios resulting in a release of radioactive materials. NUREG-0706 did identify situations where workers could receive a dose from these scenarios (but still within the Part 20 limits); the report did not identify 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 staff analyses of in situ leach uranium extraction facility operations and accidents that consider both likelihood of occurrence and consequence. The analyses in NUREG–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. The staff considers analyses similar to, or based on, those in NUREG–6733 to be an appropriate basis for licensee safety analyses.

NUREG–1569 (Standard Review Plan for In Situ Leach Uranium Extraction License Applications) did not intend to require applicants to prepare complex accident analyses, consequence evaluations, and probability determinations. However, site-specific conditions and circumstances must be addressed in any UR 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 identified situations where workers may be subjected to higher than anticipated doses, but the doses would remain within the 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 impacts 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 were reviewed to examine geologic features, structures such as surface impoundments, diversion channels, monitoring wells, and recovery plant buildings.

Staff also considered the meteorology through a description of the general climate of the region and local meteorological conditions based on appropriate data from ‘National Weather Service,’ military, or other stations recognized as standard installations. 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 impacts are substantiated as being representative of expected long-term conditions at and near the site.

Staff also considered the seismicity and the seismic history of the region included. 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.
In summary, to review and grant a uranium recovery license, site suitability features must be addressed. 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 may impact compliance with the uranium recovery facility safety objectives.

II. Post-Fukushima Event Evaluations/Assessments

The 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 flood, seismic or external fire hazards impacts would be anticipated to cause no significant impact to members of the public beyond the licensing dose criteria required in NRC regulations or guidance. For ISR facilities, the process of uranium extraction is conducted in subsurface. Therefore, potential impacts from seismic or flood hazards would be negligible, not exceeding public dose limits for normal operation or external severe conditions. For mill tailing facilities, such tailings are placed in piles at or below surface and stabilized with thick clay covers and heavy rip-rap boulders of to minimize erosion and to stabilize materials under severe conditions. Considering mill tailings pile structure, as required in NRC regulations and guidance, impacts from external events should not exceed public dose limits. Under severe external events conditions, potential dose impacts to members of the public should not exceed EPA protective action guidelines under emergency situation. In summary, for all licensed UR and mill tailings facilities, staff believes under extreme external events or conditions, dose impacts would not exceed regulatory criteria for members of the public under an emergency situation.

External Event	Outcome	Potential Risk L M H	Assessment
Flood	• Challenge to UR structures, impoundments, storage, 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	Low	External events are addressed in the license application. Therefore, severe flooding is unlikely to occur or to cause significant harm to UR facilities. Nevertheless, severe flooding may case low-activity releases from UR' impoundments where the risk consequence would be low. It is noted that because of the location of the siting requirements of UR facility, the requirements for design, drainage system, and the engineered and natural barrier systems at the facility, there should be no significant safety impacts.

Seismic	• Challenge to UR structures and integrity of containers. • Challenges to vaults and covers to contain UR impoundments and byproduct materials waste. • Challenges to integrity of process streams and potential radon release. • Challenges to the buried piping and infrastructure due to lixiviant leaks; • Challenges to integrity of chemical treatment system and potential environmental damage. • Challenges to integrity of yellowcake	Low	Engineered barriers and system design should minimize significant impacts and provide additional defense-in-depth for safety. Risk consequences under extreme conditions (e.g.; from (i) radon releases from process streams; (ii) yellowcake dryer explosions; (iii) lixiviate leaks in buried piping between the well fields and the processing facility; and (iv) chemical accidents) should be less than EPA PAGs.
High Wind and Missiles	Challenge to structures in which UR equipment and engineered systems are housed. Radon releases as well as low-activity uranium and chemicals may be release.	Low	
Lightning	Challenge to structures that may catch fire during lightning which may contain stored yellowcake which is pyrophoric.	Low	
Snow and Ice Loads	Challenge to structures in which ore is being uranium ore is being processed for concentration, calcination, and drying; and to byproduct waste in a transition before disposal.	Low	
Drought	None	Low	
Temperat-ure Extremes	None in the cold regions. In the hot and dry regions concerns of fire from pyrophoric uranium oxide, yellow cake, stored before shipment.		
External Fire	Challenge to structures in which UR plant and equipment are housed. Impacts on uranium oxide, yellow cake, stored before shipment could be significant. Pyrophoric nature of yellowcake could cause significant hazard. .	Low	
Loss of Power	None, except for possible release of radon	Low	

The NTTF report, ‘Recommendations for Enhancing Reactor Safety in the 21st Century,’ provided 12 recommendations. These recommendations have been considered for applicability to uranium recovery and mill tailing facilities and evaluated to determine if any future action is warranted. The following table summarizes the result of the review.

Table 2. 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
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).	Not Action
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
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-basis external events.	Not Applicable
5	The Task Force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable
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
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.	Not Applicable
9	The Task Force recommends that the NRC require that facility emergency plans address prolonged SBO and multiunit events.	Not Applicable
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.	Not Applicable
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
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

IV. 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 to harm workers or the public or cause any significant damage to the environment. Potential hazards from external events are believed to be insignificant or low for all above facilities. Therefore, 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 approximately 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 facilities, or sites, present low hazard/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/subsurface soil. Low hazard may also be associated with radioactive waste materials resulting from cleanup, decontamination, and dismantlement which 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.

<u>Licensing</u>

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 approximately 49 sites in Agreement States undergoing decommissioning (36 materials / 13 uranium recovery). Current NRC decommissioning sites include: (a) 17 power reactors, (b) 8 research and test reactors; (c) 16 complex non-reactor facilities; (d) 39 uranium recovery facilities; and (e) 2 fuel cycle facilities.

Regulations and Guidance

The following regulations specifically apply to facilities undergoing decommissioning:

- 10 CFR 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,
- 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 72.218 and 72.54 (Independent Storage of Spent Nuclear Fuel, HLW, and Reactor Related GTCC: (a) Termination of License; (b) Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings, and Outdoor Areas)

The primary decommissioning guidance documents are the Consolidated Decommissioning Guidance, Volumes I-IV (NUREG-1757), Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans (NUREG-1700, Rev. 1), Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NUREG-1575), and Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors (NUREG-1537). These NUREGs describe: (1) methods acceptable to the NRC staff in implementing specific parts of the Commission's regulations; (2) techniques and criteria used by the 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 for decommissioning 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 describes the information (subject matter and level of detail) needed to terminate a license by considering the specific circumstances of the wide range of radioactive materials users licensed by NRC. Volume 1 is intended to be applicable only 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.

For power reactors, a license termination plan (LTP), must be submitted two years or more prior to 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 must 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. A licensee might submit the LTP concurrently with the post-shutdown decommissioning activities report (PSDAR). Guidance on the content of the PSDAR can be found in Regulatory Guide 1.185 (Standard Format and Content Guide for Post-Shutdown Decommissioning Activities Report).

Inspection and Oversight

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

- ensure, through the review and approval of decommissioning plans 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 approved decommissioning plan and NRC's regulations;
- ensure that corrective actions are taken by facilities undergoing decommissioning if they are not conforming to the approved decommissioning plan or NRC's regulations; and
- 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 is described in Inspection Manual Chapter 2602, "Decommissioning Oversight and Inspection Program for Fuel Cycle Facilities and Material Licensees", IMC 2545, and IMC 2561 "Decommissioning Power Reactors Inspection Program" which contain the objectives and procedures to use for each type of inspection.

Emergency Preparedness

Emergency preparedness considerations for dismantlement and decontamination of facilities are conducted in the context 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 in order to assess and evaluate additional safety issues and hazards. Under such conditions, site-specific emergency preparedness would be developed. In this context, EPA's protective action guidelines would be used in the early and intermediate phases of emergency. It is noted that 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. Such exemptions are not directly related to emergency preparedness under potential external events. These exemption requests are evaluated on a case-by-case basis.

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 nuclear facility or site, such as fuel depots or areas for the storage of hazardous materials handled during decommissioning. Significant events, as either design-basis external human induced events or design-basis external natural events, are typically identified and evaluated particularly in the preliminary phases of the site evaluation process.

The decommissioning plan (DP) typically addresses meteorology, climate/climatology, 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 license's material as well as decommissioning material not in storage (see NUREG-1757 Appendix H). Further, the DP includes audit and surveillance. Notification, as well as the use of the master inspection plan; including radiation criticality safety, are significant aspects of the decommissioning reviews. External events such as flooding, earthquakes, tornados, volcanic eruptions, or severe climate conditions are usually addressed in the design of the facility and in the decommissioning plan. In the unlikely event that such external events were to occur, the status of the site is reported and a site-specific assessment of any significant safety impacts is conducted to identify immediate actions that need to be taken to mitigate any 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, in order 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 Subpart E of this part.

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

II. Evaluations and Assessments Prior to Fukushima

Prior to Fukushima, the staff evaluated the potential impacts of external events on certain nuclear power reactor facilities that ceased operation and experienced a transition before entering into a formal decommissioning mode (e.g. before submission of a license termination plan, such as reactors going through a long-term "safe-store"). The concern was for facilities where defueling or removal of spent fuel has not yet been completed. In this context, two analysis and studies were conducted for spent fuel pools under emergency or external events situations. NUREG-1738 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 provide a 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 spent fuel pool at decommissioning plants; the risk assessment of spent fuel pool accidents; the consequence calculations; and the sensitivity study and implications for decommissioning regulatory requirements. NUREG-1738 concluded that: "given the robust structural design of Spent Fuel Pools (SFP), it is expected that a seismic event with peak spectral acceleration several times larger than the safe shutdown earthquake (SSE) 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 (SECY-13-0029). 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.

Another study which was conducted by NRC staff in 2005 focused on spent fuel pools at: Zion 1 and 2, Millstone Power Station, Unit 1; La Cross Boiling Water Reactor, Humboldt Bay 3, Indian Point Nuclear Generating Unit No. 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 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 6 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."

III. Post Fukushima Event Evaluations/Assessments

On December 30, 2014, (SRM –SECY-14-0118) (ADAMS Accession No. ML14364A111) the Commission directed staff to proceed with rulemaking on decommissioning. This rulemaking will address issues discussed in SECY-00-0145, such as: the graded approach to emergency preparedness; lessons learned from the plants that have already (or are currently) going through the decommissioning process; the advisability of requiring a licensee's Post-Shutdown Decommissioning Activity Report (PSDAR) to be approved by the NRC; the appropriateness of maintaining the three existing options for decommissioning and the timeframe associated with those options; the appropriate role of state and local governments and non-governmental stakeholders in the decommissioning process; and any other issues deemed relevant by the NRC staff. The Commission directed staff to set an objective of early 2019 for completion of this rulemaking. 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 details in the context of the anticipated rulemaking for decommissioning. 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.

External Event	Outcome	Potential Risk L M H	Assessment
Flood	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.	Low	During the decommissioning process the licensee should 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, the licensee should assess site location, climate conditions, and potential safety issues related to certain events such extreme rain or flooding conditions, tornados, and high wind events. It is noted that 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 staff does not anticipate 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). Emergency plans that address natural phenomena, including an earthquake, a tornado, flooding, or other phenomena, generally are not required for decommissioning facilities though addressed in safety guides.
Seismic	Challenge to structures in which radioactive materials are stored or present as residual radioactivity above release limits.	**Low**	
High Wind and Missiles	Challenge to structures in which radioactive material is present above release limits or stored in a transition for disposal.	**Low**	
Lightning	Challenge to structures that may catch fire during lightning which may contain stored radioactive materials in a transition before disposition.	**Low**	
Snow and Ice Loads	Challenge to structures in which radioactive material is stored or being treated for disposition.	**Low**	
Drought	None	**Low**	
Temperature Extremes	None except for pyrophoric materials stored before disposition.	**Low**	
External Fire	Challenge to structures in radioactive materials stored in a transition for disposition.	**Low**	
Loss of Power	none	**Low**	

The NTTF report, 'Recommendations for Enhancing Reactor Safety in the 21st Century,' provided 12 recommendations. These recommendations have been considered for applicability to decommissioning facilities and evaluated to determine if any future action is warranted. The following table summarizes the result of the review.

Table 2. 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
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).	Not Action
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
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-basis external events.	No Action
5	The Task Force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	No Action
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
7	The Task Force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	No Action
8	The Task Force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs.	No Action
9	The Task Force recommends that the NRC require that facility emergency plans address prolonged SBO and multiunit events.	No Action
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
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
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

IV. Conclusion and Recommendations

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. NRC staff concludes that facilities or sites undergoing formal decommissioning are licensed, as appropriate, to the scope and potential hazard created. Potential hazards from external events are believed to be insignificant or low. Therefore, no further study or regulatory action is recommended for such facilities. Previous studies and analysis (e.g. NUREG-1738, and NRC Staff Study conducted on November 16, 2005), showed that certain facilities that ceased operation and experienced a transition before entering into a formal decommissioning mode (e.g. before submission of a license termination plan, such as reactors going through a long-term "safe-store") need no further analysis. Such facilities include those where defueling or removal of spent fuel has not been completed yet. In addition, as directed by the Commission in SRM –SECY-14-0118, regarding the graded approach to emergency preparedness, more detailed analysis to support the rulemaking would be anticipated to alleviate any concerns regarding emergency issues and external events.

8. NRC-Licensed Non-power Reactors Updated on 3/2/15

I. Introduction

The 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 the 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 due to their minimal decay heat generation which only needs air cooling to adequately remove sufficient 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 due to their reliance on water for adequate decay heat removal. The 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.

II. Background

Non-power reactors (NPRs) are designed for mulitiple purposes including commercial, research, testing and education. These reactors are not used to produce electricity, process steam, or for desalination. Currently, 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 currently licensed by the NRC, 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 will be assessed by the Office of *Nuclear Materials Safety and Safeguards.*

The predominant radiological hazard associated with RTRs while operating within its design basis is radiological exposure from the mishandling of radioactive materials or experiments. However, when one considers the radiological hazards under postulated beyond design basis conditions, it is necessary to consider the release of fission products. These radioactive materials are normally contained within the fuel cladding. In most cases, the reactor fuel is located inside the core which maintained within pools of water for purpose of shielding personnel from radiation and the removal of heat produced from nuclear reactions. The reactors are housed within confinement or containment buildings. Therefore, an uncontrolled release of radioactive material could only happen 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, and pressures; and duration and frequency of operation. These operating characteristics result in significantly larger fission product inventories for the power reactors due to the significantly larger quantity of fuel in both the core and spent fuel pool. To put these differences into perspective, consider the following: 1) the maximum licensed thermal power level of the smallest power reactor is approximately two orders of magnitude greater than that of the largest RTR; 2) the duration of operations of a power reactor is nearly continuous, while the majority of RTRs are only operated periodically, typically a fraction of the normal work week; 3) the onsite accumulation of significant quantities of spent fuel (20 years' 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 which can consist of both power and NPRs. Section 104 of the Act pertains 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. Paragraph 104c 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 a 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.

Additionally, 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 events, up to and including design-basis accidents (DBAs). 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 implemented. 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.

Unique to the licensing of RTRs, is the analysis of a maximum hypothetical accident (MHA). The MHA assumes a release of fission products from the reactor’s fuel or fueled experiment. Analysis of the MHA is necessary because unlike power reactors, most RTRs are designed and operated so that the DBAs are not postulated to involve a radioactive release. 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. Since the MHA is not expected to occur, only the potential consequences are analyzed and not the initiating event and 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 research reactors are based on meeting conservative assurance that the public and occupational dose resulting from the postulated radiological release remain bounded by the normal (non-accident) occupational and public dose limits contained in 10 CFR Part 20, “Standards for Protection Against Radiation.” In a very few instances, the staff has approved licensing actions for research reactors that exceed the occupational and public dose limits of Part 20 when supported by highly conservative accident analyses. For test reactors, licensing criteria allowing construction and permitting operation is based on meeting 10 CFR Part 100, “Reactor Site Criteria.”

The impacts of external events are also specifically considered in the RTR licensing process. RTRs licensed by the NRC are required to demonstrate, in their design basis, reasonable assurance that external events would not preclude 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. Additional inspections, beyond those required by the routine inspection program, are completed as needed (e.g., in response to an event).

Those reactors that are shutdown and are not being decommissioned have an abbreviated inspection program completed triennially. NRC inspects decommissioning reactors to verify their safe condition to 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[30] 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,26 of RTRs (those less than 2 MW_t), the EPZ is the operational boundary[31] and the EPZ of the highest thermal power NRC-licensed RTR (20 MW_t) is 400 meter radius surrounding the facility.

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

RTRs also implement 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 currently licensed because the postulated radiological consequences of the MHA at each RTR does not meet the criteria requiring a General Emergency[32] declaration at the site boundary[33] (off-site).

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

The NRC RTR staff has participated with our international partners in gaining 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. This work was deemed necessary because the stress test methodology initially used in Western European post-Fukushima assessments was designed specifically for power reactors. The methodology was applied to some of the European research reactors with mixed results. Based on presentations at technical meetings from countries that applied the methodology to their research reactors, the stress test methodology was found to be much more effective for the higher powered research reactors. Application of the stress test methodology to lower powered research reactors resulted in little, if any, useful information but required the application of significant resources to complete. [34]

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" 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 reactor(s). A few countries (mostly Western European) have met the intent of the IAEA guidance. The 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.

[32] 10 CFR Part 50 Appendix E

[33] **Site boundary:** The site boundary is that boundary, not necessarily having restrictive barriers, surrounding the operations boundary wherein the reactor administrator may directly initiate emergency activities.

[34] European Research Reactor Conference 2012.

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

As a result 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 the 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 current regulatory framework and to identify any necessary changes to optimize NRC, licensee, local, State, and Federal response capabilities. In the implementation of this directive, the 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 a number of theft and sabotage scenarios at each RTR. These scenarios were chosen as they were viewed as having the potential to adversely affect to 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. The staff employed a three-phase, assessment process that had been approved by the Commission[35]. 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 current 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, provide useful insights to assist in the assessment of the consequences associated with a beyond-design-basis external event since both may result in severe damage states to the facility that may exceed those assumed in RTR's design and accident analysis. The post-9/11security assessments considered the potential radiological consequences for the various sabotage scenarios. The supporting analyses of the radiological consequences demonstrated that for all RTRs, sabotage scenarios would result in doses to the public that are fractions of the 10 CFR Part 100 reactor siting dose criteria[36]. Therefore, the staff has considered the sabotage event conclusions when assessing the beyond-design-basis external events because the extent of facility damage assumed in the worst case (limiting) sabotage scenario is expected to bound the consequences of all the beyond-design-basis external events.

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

[36] 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.

VIII. Prompt Post-Fukushima Assessments

In the days immediately following the Fukushima accident, the NRC staff collected available information related to accident initiation, progression, and consequences. This information was used by the RTR staff 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. Staff conclusions were based on the available information and engineering judgment. The areas considered by the staff during the prompt assessment included:

- Natural Events
- 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 the NRC staff to compare the site’s potential natural hazards to the applicants proposed facility design. This allows the staff to reach a conclusion on the acceptability of the proposed design at that site. In order 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 following postulated natural events.

Following the Fukushima Dai-ichi accident, the staff reviewed the natural event descriptions and discussions provided by each RTR licensee in licensing documentation. The first of these assessments was performed by staff the day of the accident. The staff specifically assessed the designs of those RTRs located 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 located in Pacific States were vulnerable to the tsunami when reaching the locations of the facilities.

In the weeks that followed the accident, the staff initiated 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. The 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 considered 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 meets accident analysis acceptance criteria which have been based on highly conservative public and occupational dose limits.
- 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

On March 11, 2011, the magnitude 9.0 earthquake caused the automatic shutdown of all the operating units at the Fukushima Dai-ichi site and resulted in the loss of off-site power to all six units. The emergency diesel generators at all the six units started and were providing emergency alternating current (AC) power for decay heat removal and other critical systems. Approximately 40 to 50 minutes following the earthquake, the site experienced a tsunami initiated by the earthquake. The tsunami height exceeded the site's designed tsunami protection by an estimated 27 feet. The inundation of sea water caused extensive damage to important plant structures, systems and components resulting in the loss of emergency AC power to five of the six units, a condition known as a station blackout.

Given the Fukushima scenario, the staff included a review of an extended loss electrical power on the safety of NRC-licensed RTRs as part of their immediate assessment. The 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 shutdown the reactor.
- 28-out-of-31 RTRs are air-coolable and thus do not require electric power for decay heat removal..
- 3-out-of-31 RTRs may require AC power to replenish inventory lost because of seismic activity or boil-off.

Decay Heat Removal

At Fukushima it was ultimately the inability to remove the decay heat[37] 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

[37] 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 decrease 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

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[38].

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

- 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 2 MWs (26 of the 31 RTRs)
- Loss of coolant scenarios in the 2 megawatt and greater reactors (5 of the 31 RTRs) rely on designs that:
    - maintain floodable volume that maintains the core covered with water at three of those facilities
    - provide the capability to spray the core with diverse sources of water via active (pumped) and/or passive (gravity drain)
    - are equipped with emergency power systems or batteries sufficient to power cooling and makeup systems for the limited time required following reactor shutdown

Spent Fuel Pool Cooling

Given the public interest in spent fuel pools following the Fukushima accident, the staff included a review of the storage of irradiated and spent fuel at RTRs. The 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 2MW 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

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

- 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, the staff included a review of hydrogen generation and control at RTRs. The 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 have dedicated hydrogen control systems 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
- 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 time 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 as a result of the inability 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, the staff conducted a review of the containment and confinement structures used at RTRs to prevent or control the release radioactive materials to the environment. The 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 as a result of normal operating, transient or accident conditions

Prompt Assessment Conclusions

The RTR prompt assessment was completed within weeks of the Fukushima Dai-ichi accident. As discussed above, the 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 following the Fukushima Dai-ichi accident, the Commission directed the staff to conduct a systematic and methodical review of the NRC's processes and regulations to determine whether the NRC should make additional 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 provided 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. The 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, the NRC staff chose to first group the RTRs based on their licensed thermal power level, since 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.

The use of a 2 MW_t threshold for categorizing RTRs was not arbitrary. A number of analyses[39] exist that demonstrate that air cooling is sufficient to remove decay heat from research reactors licensed for operation at less than 2 MW_t following a complete loss of coolant (dry core). One analysis of research reactor plate-type fuel[40] (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 additional 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.

<u>Assessment of External Events on RTRs</u>

The staff conducted an assessment of 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 (DBA and MHA) and were considered by the staff during the licensing process. 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.

RTRs less than 2 MW_t (Category 1 Research Reactors)

The staff found that the risk of a significant release of radioactive material as a result of a beyond design basis external event 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

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

reactors results in a low decay heat generation rate 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 radiological consequences assumed to result from the MHA are highly likely to bound the potential radiological consequences from all beyond design basis external events for the Category 1 research reactors.

Table 1 below summarizes the external events reviewed by the staff, the impact on the facility and a summary of the basis for no additional action being needed for this class of RTRs.

Table 1

External Event	Potential Impact 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)[41] 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 historically meet the local building codes for the USGS seismic zone for their location. • Building code bases is to prevent building collapse. • Damage from building collapse bounded by existing analysis. • Loss of air cooling from debris obstruction highly unlikely. • Fuel cladding remains intact. • No radiological release expected.
High Wind/Tornado/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

[41] 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 Impact on the Facility	Assessment
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/ 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 confinement due to loss of ventilation systems	• Bounded by Flood and Seismic Assessment above
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 on maximum licensed thermal power. Given the potential for increased decay heat removal requirements for reactors in this category, the 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 to a beyond design basis external event similarly to that of the research reactors in Category 1. 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 (0.12 MW_t and 0.81 MW_t) produce less decay heat than for a 1 MW_t research reactor. Given the fractional utilization of their maximum licensed power over the life of these facilities, it can be demonstrated that decay heat can be adequately removed 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 from their MHAs are bounding for all beyond design basis external events. Therefore, the staff considers the analysis summarized in Table 1 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. The staff 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

The two high powered research reactors in Category 2 are both tank type reactors capable of removing adequate decay heat by the natural convection flow of the reactor coolant following a beyond design basis 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 credible external event that could result in conditions were there is a LOCA and concurrent loss of ac power is a beyond design basis seismic event. A seismic event would be required to significantly exceed 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 6 MW_t research reactor is located 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 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 for the 6 MW_t has concluded that the seismically induced loss-of-coolant accident is not credible. Considering the similarity between the designs of the 6 and 10 MW_t research reactors, 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 the research reactors are such that a LOCA induced from a seismic event is not credible. Furthermore, 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.

The 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 reactors are below grade or protected by above grade reinforced concrete biological shield that provide additional barriers (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, the 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 the staff will consider the results of the previous security assessments above before additional regulatory actions, if any, are considered. As part of their assessment of these events, the staff intends to gather the appropriate structural expertise, supplemented by staff that have performed assessments of power reactor responses to the mitigating strategies Order to inform its assessment on whether additional regulatory action is needed. Table 2 below summarizes the external events reviewed by the staff, the impact on the facility and the staff's assessment.

Table 2

External Event	**Potential Impact 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)	• Research reactors historically meet local building codes for the USGS seismic zone for their location. • The building code 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. Staff to continue assessment.

External Event	Potential Impact on the Facility	Assessment
High Wind/Tornado/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 pool liner and fuel. • Staff to continue assessment to ensure that there is not an event-specific vulnerability
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 • 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.	• 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 MWt Test Reactor

The 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 emergency cooling tank will make-up the inventory boiled-off during the first 0.5 hour. The heavy water storage 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, 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. In severe accidents (beyond design basis accidents), domestic light water can be supplied to the reactor through a spool-piece and double isolation valves. Emergency power to the active decay heat removal system can be supplied by the 125 VDC 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, there are arrangements to collect the heavy water in sumps and pump the water 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 beyond design basis external event may exceed those assumed in the MHA but would not likely exceed 10 CFR 100 siting criteria used in licensing the facility. This has been confirmed by the post-September 11th 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 the10 CFR Part 100 reactor siting dose criteria.

Nevertheless, the 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 the 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. There are no major bodies of water near to the facility and the facility is located above the 500 year flood plain. Since there are no major sources of water in proximity to the facility, there are also 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. Flooding from local intense precipitation (LIP) events is extremely rare as demonstrated by historical precipitation records. 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. The staff plans to continue its assessment of flooding events to determine if additional regulatory actions are needed to address this event.

The 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 the staff intends to gather the appropriate structural and flooding expertise, supplemented by staff that have performed assessments of power reactor responses to the mitigating strategies Order to inform its assessment on whether additional regulatory action is needed. Table 3 below summarizes the external events reviewed by the staff, the impact on the facility and the staff's assessment.

Table 3

External Event	Impact 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.
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 seismic criteria. • The test reactor decay heat is adequately removed by natural convection of pool water for at least 0.5 hours. Continued adequate decay heat removal will require either a source of makeup water or active decay heat removal. • Staff will continue assessment to ensure there is not an event-specific vulnerability
High Wind/Tornad o/Missiles	• Loss of all electrical power and active heat removal systems • Challenge to reactor structures including confinements or containments	• Reinforced concrete biological shield prevents impact damage to pool liner and fuel. • 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. • Staff will continue assessment to ensure there is not an event-specific vulnerability
Lightning	• Loss of all electrical power and active heat removal systems	• 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. • 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 pool water initially. Continued adequate decay heat removal will require active decay heat removal systems in the long-term. • 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.

External Event	Impact on the Facility	Assessment
		• Long-term coolant makeup requirements very minimal for this reactor because of low decay heat. • 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. • Below grade design or above grade reinforced concrete biological shield prevents direct heating of the pool liner and fuel • 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. • 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 pool water initially. Continued adequate decay heat removal will require active decay heat removal systems in the long-term. • 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 pool water initially. Continued adequate decay heat removal will require active decay heat removal systems in the long-term. • Staff will continue assessment to ensure there is not an event-specific vulnerability

Assessment of NTTF Recommendations

The Fukushima Dai-ichi accident resulted from an external event that exceeded the design of the facility. The external event overwhelmed all levels of the facility's defense-in-depth eliminating plant equipment and barriers relied upon to prevent the release of radioactive material as well as presenting significant challenges to their emergency preparedness preparations.

The NTTF focused on the Fukushima Dai-ichi accident lessons learned and made recommendations to enhance the defense-in-depth associated with U.S. power reactors. The RTR staff conducted an assessment of the 12 NTTF recommendations and considered the application of those recommendations to the NRC licensed RTRs. The 12 recommendations are grouped into 5 categories:

- clarifying the regulatory framework
- ensuring protection
- enhancing mitigation
- strengthening emergency preparedness
- improving the efficiency of NRC programs

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 cool 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 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 test reactor may 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 recommendations may prove beneficial under rare circumstances in preventing fuel cladding failure at the test reactor. As discussed in the section above regarding external events, the staff is continuing its assessment of the need for protective or mitigative strategies for these reactors. If the staff concludes that additional protective or mitigative strategies are appropriate the staff will use existing regulatory processes to implement them.

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 cool 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 the majority 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. The staff has not identified additional actions needed at this time for the two 2 MW_t research reactors in Category 2. As noted above, the staff will continue to evaluate possible improvements to the regulation of this class of research reactors as work on the risk management regulatory framework initiative progresses.

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. Following 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 the 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 the staff will also assess the capability of other external events (e.g., missiles from high winds) causing an extended loss of electrical power. 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 the 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 to as criteria to justify that no additional security-related regulatory actions were necessary to fulfill Commission's obligations under the Atomic Energy Act[42] 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 is highly likely to bound the facility damage of beyond-design-basis external events. Considering the bounding nature of the worst case sabotage scenario, it is reasonable to conclude that the radiological consequences resulting from any beyond design basis external event will likely not exceed the test reactor licensing criteria.

As discussed above, for the test reactor, the staff is performing additional assessments of the resiliency of the facility's emergency power and active decay heat removal systems to a beyond design basis seismic and external events. If the additional assessment demonstrates that the emergency power and active decay heat removal systems remain capable to perform their functions following such events, the 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 beyond design basis seismic or external event, then staff will use the appropriate regulatory process to implement additional features to prevent or mitigate these beyond design basis seismic and external events at the test reactor.

The staff believes it is reasonable to complete the additional assessment of the three greater than 2 MW_t reactors discussed in the previous paragraphs within the subsequent 24 months.

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[42] 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.