# Assessment of Lessons Learned from the Fukushima Dai-ichi Nuclear Accident to Research and Test Reactors in the United States

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The United States Nuclear Regulatory Commission (NRC) staff evaluated the thirty-one NRC-licensed research and test reactors (RTRs) to assess the applicability of lessons learned from the Fukushima Dai-ichi nuclear accident to these facilities. The NRC staff assessed two categories of RTRs based on the licensed thermal power level of each facility. Category 1 consisted of the twenty-six research reactors licensed to operate at power levels lower than 2 megawatts-thermal ($MW_t$). The remaining five NRC-licensed RTRs licensed to operate at 2 $MW_t$ or higher comprised Category 2. The NRC staff's assessment concluded that all of the Category 1 and the two lowest-powered Category 2 research reactors were highly resilient to the loss of electrical power, active decay heat removal systems, and heat sink. These twenty-eight research reactors generate minimal decay heat, which can be adequately removed via air cooling to prevent fuel cladding failure. In contrast, the three largest Category 2 RTRs rely on the availability of water for adequate decay heat removal. As such, the NRC staff performed an additional assessment of these three facilities to determine the resilience of their primary coolant systems to a beyond-design-basis seismic event. For the 20 $MW_t$ test reactor, the NRC staff also assessed the resilience of emergency power, active decay heat removal systems, and coolant make-up systems for flooding and beyond-design-basis seismic event. The results of the NRC staff's assessment revealed that the existing design-bases for NRC-licensed RTRs adequately protect against fuel cladding failures and the release of radioactive material during a beyond-design-basis external event.

## 1. Introduction to Research and Test Reactor Licensing Methodology

All thirty-one currently operating non-power reactors (NPRs) in the United States (U.S.) are research and test reactors (RTRs) designed and used for research, testing, and education purposes. Unlike commercial nuclear reactors, these facilities are not used to produce electricity or process steam. The licensed power levels of these facilities varies from 5 watts-thermal to 20 megawatts-thermal ($MW_t$).

Research and test reactors are licensed using the concept of defense-in-depth. This approach to licensing these facilities compensates for uncertainties in reactor design, operation, and radiological consequences associated with potential accidents. Further, RTRs have been historically licensed using deterministic analysis methods, including highly conservative safety margins. Guidelines for licensing RTRs is provided in, NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" [1, 2].

As part of its licensing evaluations, the U.S. Nuclear Regulatory Commission (NRC or Commission) staff assesses the risk to workers and the public health and safety resulting from operation of RTRs. The predominant radiological hazard to workers at these facilities is radiological exposure resulting from the mishandling of radioactive materials or experiments by reactor operators or researchers. The greatest radiological hazard to members of the public is the accumulation of fission products within uranium fuel and fuel cladding. However, the reactors and fuel are housed within confinement or containment buildings to prevent and mitigate uncontrolled releases of radioactive material.

In order to ensure conservatism in facility design and protection from radiological hazards, a set of licensing-basis events have been established to cover a wide spectrum of postulated accidents at RTRs. It is common that the analysis of a set of postulated credible accidents at an RTR does not result in a radiological release. Therefore, in order to assess the potential dose impact of a facility to the public, an incredible hypothetical radiological fission product release is analyzed. This event, referred to as the maximum hypothetical accident (MHA), must bound all the credible hazards resulting from the postulated accidents. For currently licensed RTRs, the MHA assumes a failure of the fuel or a fueled experiment that results in radiological consequences that exceed those of credible accidents. Since the MHA is not expected to occur, only the potential consequences of the event are analyzed and not the initiating event or scenario details.

For research reactors, the MHA is to be based on conservative assumptions such that radiological consequences from such an event are bounded by the occupational and public dose limits in Title 10 of the *Code of* Federal Regulations (10 CFR) Part 20, "Standards for protection against radiation." For test reactors, the public and occupational doses resulting from the postulated MHA release must meet the siting and accident dose criteria in 10 CFR Part 100, "Reactor site criteria."

The NRC also considers the effects of external events, including seismic and flooding events, on RTRs as part of its licensing process. At a minimum, RTRs are designed to meet local building codes to demonstrate reasonable assurance that external events would not prevent the safe operation and shutdown of the reactor. The radiological consequences of external events are considered in the facility's analyzed accidents and bounded by the MHA.

The radiological consequences resulting from potential accidents at currently operating RTRs are significantly less than those associated with nuclear power reactors due to the differences in operating characteristics between these types of facilities. For example, RTRs operate at temperatures below the boiling point of water at atmospheric pressure and at or near atmospheric pressure, which limits the energy that must be controlled or dissipated during an accident. Factors such as lower operating power levels, temperatures, and pressures; reduced hours of operation; and smaller quantities of onsite fuel (including spent fuel) contribute to significantly smaller nuclear material inventories, accident source terms, and demand for active decay heat removal at RTRs compared to nuclear power reactors.

## 2. Prompt Post-Fukushima Dai-ichi Assessments at Research and Test Reactors

In the days immediately following the Fukushima Dai-ichi accident, the NRC staff collected available information related to the accident's initiation, progression, and consequences. The NRC's RTR staff used this information to inform a prompt assessment of the safety of NRC-licensed RTRs. The NRC staff documented the results of its prompt assessment in a 2015 Commission paper that evaluated the applicability of lessons learned from the Fukushima Dai-ichi accident to facilities other than operating nuclear power reactors [3]. The prompt assessment, as described below, was used to determine whether any conditions at NRC-licensed RTRs could adversely affect the public health and safety and the need for immediate regulatory action. The areas considered by the NRC staff as part of the prompt assessment included natural events, electrical power, decay heat removal, spent fuel, combustible gas control, and reactor containment and confinement. The RTR prompt assessment was completed within weeks of the Fukushima Dai-ichi accident. As described below, the NRC staff determined that there were no safety concerns revealed by the accident for which immediate actions were necessary at NRC-licensed RTRs, nor was any new information revealed that would contradict or invalidate the licensing basis for any of these facilities.

**2.1. Natural Events**

As part of the NRC's licensing process, applicants for RTR licenses provide information on the geographical, geological, seismological, hydrological, and meteorological characteristics of the reactor site. As part of its evaluation of this information, the NRC staff determines whether the design of the facility provides reasonable assurance that structures, systems, and components will remain capable of performing any necessary safety functions during and after postulated natural events.

After the Fukushima Dai-ichi accident, the NRC staff reviewed the natural event descriptions and discussions for each NRC-licensed RTR. Initially, the NRC staff assessments focused on the designs of those RTRs in the Pacific Coastal States in light of predicted tsunami forecasts. However, the NRC staff concluded that none of the five RTR sites in the Pacific Coastal States were vulnerable to a tsunami. In the weeks following the accident, the NRC staff initiated a broader assessment of natural events considered in the design bases for NRC-licensed RTRs. A specific emphasis was placed on seismic and flooding events. Based on its assessment of these events, the NRC staff concluded the following related to the current design and siting of NRC-licensed RTRs:

- The average and extreme natural event conditions are considered in the assessment of potential facility hazards during licensing.

- Due to the low-power operation of NRC-licensed RTRs, these facilities do not rely on large sources of water for makeup and heat sinks. This allows RTR facilities to be sited at significant distances from potential sources of flooding.

- The radiological consequences associated with the maximum predicted seismic event at each research reactor site meets the accident analysis acceptance criteria in NUREG-1537, which is based on the conservative public and occupational dose limits in 10 CFR Part 20. The one NRC-licensed test reactor meets the 10 CFR Part 100 seismic requirements and radiological limits.

**2.2. Electrical Power**

The NRC staff reviewed the impact of an extended loss of electrical power on the safety of NRC-licensed RTRs as part of its prompt assessment to determine each facility's reliance on and sensitivity to the loss of electrical power. Based on its assessment the NRC staff concluded that most RTRs include emergency power for equipment such as area radiation monitors, evacuation alarms and lighting, and security systems. However, none of these facilities require electrical power to safely shut down the reactor. Additionally, 28-out-of-31 RTRs can be adequately cooled by air convection and do not require electrical power for decay heat removal. Only two NRC-licensed RTRs may require alternating current (AC) power to replenish reactor coolant inventory lost due to a seismically-induced failure of the reactor coolant system boundary.

**2.3. Decay Heat Removal**

Due to the low operating power, short duration of operation, and limited quantities of irradiated fuel onsite, RTRs generally do not require complex, diverse, and redundant active decay heat removal systems typically found at nuclear power reactors. For example, the decay heat produced at the twenty-six NRC-licensed RTRs operating at power levels lower than 2 $MW_t$ can be adequately removed through air cooling of the core [4]. As such, these facilities are less susceptible to core damage from overheating compared to nuclear power reactors. For five of the RTRs operating at greater-than-or-equal-to 2 $MW_t$, loss of coolant scenarios rely on designs that maintain core coverage with water, provide core spray capabilities, or are equipped with emergency power or batteries sufficient to power necessary cooling and makeup systems for the removal of decay heat.

As part of its immediate assessment of the decay heat removal capabilities of RTRs, the NRC staff found that natural convection of the primary coolant provides adequate short-term (0.5 to 2.5 hours) decay heat removal for all NRC-licensed research reactors. Only the one NRC-licensed test reactor requires active systems for adequate long-term decay heat removal.

### 2.4. Spent Fuel Pool Cooling

All NRC-licensed RTRs have limited inventories of spent or irradiated fuel. Due to the low operating power and limited duration and frequency of operation spent fuel is not routinely discharged at twenty-eight of the NRC-licensed research reactors. Additionally, the U.S. Department of Energy recovers spent or unwanted irradiated fuel from government-owned reactors and academic institutions, which prevents the accumulation of onsite spent fuel inventories. Irradiated and excess RTR fuel that is kept onsite is typically stored dry, or in wet storage in reactor pools, or in dedicated spent fuel pools. At none of these facilities is active cooling required to adequately remove decay heat from spent or irradiated RTR fuel.

### 2.5. Combustible Gas Control

As part of its prompt assessment, the NRC staff evaluated hydrogen generation and control at RTRs. Based on its review, the NRC staff determined that the radiolytic decomposition of water at those research reactors licensed to operate at power levels lower than 2 $MW_t$ does not produce sufficient amounts hydrogen to reach combustible or explosive concentrations within the reactor building. Higher-powered RTRs either prevent the formation of combustible or explosive concentrations of hydrogen through the use of a dedicated hydrogen control system or have provided analyses demonstrating that such an environment is not credible under normal operating or accident conditions. Additionally, the limited amounts of reactive metals (e.g., zirconium, stainless steel, or aluminum) and limited decay heat do not support the conditions necessary to form hydrogen from metal-water reactions during accident conditions at NRC-licensed RTRs.

### 2.6. Reactor Confinement and Containment

The NRC staff assessed the ability of confinement and containment structures used at RTRs to prevent or mitigate the release of radioactive materials to the environment. Twenty-eight of the NRC’s licensed RTRs are designed with confinement structures. These structures are not pressure-tight and typically consist of a ventilation system used to maintain the building at a slight negative pressure. Radioactive releases are discharged through controlled, filtered, and elevated pathways within the confinement. Only three NRC-licensed research reactors use containment buildings to provide a controlled leakage boundary. Based on its review of the confinement and containment structures at NRC-licensed RTRs, the NRC staff determined that energetic releases from the reactor core or reactor coolant systems are not expected to challenge facility design limits during normal operating, transient, or accident conditions due to the low operating temperatures and pressures; large confinement and containment volumes; and limited decay heat generation rates at these reactors.

## 3. Assessment of Near-Term Task Force Report Recommendations

In the days following the Fukushima Dai-ichi accident, the Commission directed the NRC staff to determine whether improvements to the NRC's regulatory framework and processes were necessary. In response to the Commission's direction, the Near-Term Task Force (NTTF) was established to conduct a near-term evaluation of the need for agency action. On July 12, 2011, the NTTF issued its report, "Recommendations for Enhancing Reactor Safety in the 21st Century," which contained twelve recommendations related to clarifying the NRC's regulatory framework, ensuring protection, enhancing mitigation, strengthening emergency preparedness, and improving the efficiency of NRC programs [5]. While the focus of the NTTF was specific to nuclear power reactors, the NRC staff developed a review process to apply relevant report recommendations and other lessons learned from the Fukushima Dai-ichi accident to NRC licensees other than nuclear power reactors, such as RTRs.

In assessing the applicability of the lessons learned from the Fukushima Dai-ichi accident to RTRs, the NRC staff evaluated the potential impact of beyond-design-basis external events on RTRs based on their thermal power level and decay heat generation, which correlate to a reactor's susceptibility to core damage in loss-of-cooling and loss-of-power accident scenarios. As a result, the NRC staff created two review categories for RTRs. Category 1 included twenty-six of the NRC's thirty-one licensed research reactors with licensed power levels lower than 2 $MW_t$. Category 2 included the four remaining research reactors and one test reactor licensed to operate at power levels of 2 $MW_t$ or higher.

The 2 $MW_t$ threshold for the RTR assessment categories was selected based on analyses from 1982 demonstrating that air cooling sufficiently removes decay heat from research reactors operating at power levels lower than 2 $MW_t$ following a complete loss of coolant event [4]. This conclusion has been confirmed by more recent facility-specific accident and thermal hydraulic analyses performed in support of either highly-enriched uranium to low enriched uranium reactor fuel conversions or license renewal reviews for research reactors operated at power levels lower than 2 $MW_t$.

In contrast to Category 1 research reactors, it cannot be assumed that air cooling will provide adequate decay heat removal for Category 2 RTRs. Given the potential for increased decay heat generation at reactors operating at a power of 2 $MW_t$ or greater, reliance on the availability of reactor coolant, a heat sink, and electrical power becomes more important.

While the 1982 decay heat study referenced in this paper assumes the continuous operation of a reactor at 2 $MW_t$, in a more realistic consideration of power history and operation, some Category 2 research reactors would respond similarly to Category 1 research reactors during severe external events. By calculating a facility effective power (FEP), a more accurate approximation of the actual decay heat that must be removed to prevent a fuel cladding failure can be determined. The FEP for the two 2 $MW_t$ facilities produce less decay heat than a 1

$MW_t$ research reactor (i.e., 0.12 $MW_t$ and 0.81 $MW_t$). Given the fractional use of these facilities' maximum-licensed power level over the course of operation, it can be demonstrated that decay heat can be adequately removed by natural convection of pool water or by air in a loss-of-cooling event. Because of the similarities between the two 2 $MW_t$ Category 2 research reactors and Category 1 reactors, the radiological consequences postulated in the MHAs of these facilities would bound the beyond-design-basis external events considered in the sections of this paper that follow. Therefore, the NRC staff considers the analyses summarized for Category 1 research reactors to be applicable to the two 2 $MW_t$ research reactors.

The FEP of the three remaining reactors in Category 2 exceeds 2 MWt, indicating that air cooling would not be adequate to prevent a fuel cladding failure during a loss-of-cooling event.

### 3.1. Assessment of Beyond-Design-Basis External Events on Research and Test Reactors

As part of its assessment of the lessons learned from the Fukushima Dai-ichi accident on RTRs, the NRC staff evaluated various beyond-design-basis external events to determine whether more severe events than those selected as part of the licensing bases for these facilities could result in greater radiological consequences than those postulated in the accident analyses. As part of its licensing reviews for RTRs, the NRC staff considers the average and extreme historical values and predictive potentials for specific external events based on the location of a particular facility. As part of its review of beyond-design-basis external events, the NRC staff assessed the potential effects of flooding; seismic; high wind, tornado, and wind-driven missile; lightning; snow and ice load; drought and extreme temperature; fire; loss of power; and loss of heat sink events for each Category of facility described above.

### 3.2. Potential Impact of Beyond-Design-Basis External Events to Category 1 Research Reactors

Category 1 research reactors represent the majority of the NRC-licensed research reactors with a maximum licensed power level of 1.1 $MW_t$. The NRC staff found that the risk of a significant release of radioactive material resulting from an external event exceeding the severity of those events considered during licensing to be very low for the Category 1 research reactors and two lowest-powered Category 2 reactors. As discussed above, the low thermal power rating of these facilities generates limited decay heat following shut down. As such, air cooling of the fuel is sufficient to prevent overheating of the fuel cladding – even if an external event causes or happens concurrently with a complete loss of coolant, loss of electrical power, and the loss of decay heat removal systems. Table I, below, summarizes the potential effects of flooding and seismic events on the Category 1 research reactors. The effects summarized in this table are also applicable to the two lowest-powered Category 2 reactors. The NRC staff found that the potential impacts of high wind, tornado, and wind-driven missile; lightning; snow and ice load; drought and extreme temperature; fire; loss of power; and loss of heat sink events were all bounded by the flooding and seismic assessments for these facilities.

TABLE I: Effect of Flooding and Seismic Events on Category 1 Research Reactors

| External Event | Potential Effect on the Facility | Assessment |
|---|---|---|
| Flooding | • Loss of all electrical power and any active heat removal systems.<br>• 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.<br>• 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)[1] thereby reducing the susceptibility to flooding. |
| Seismic | • Loss of all electrical power and any active heat removal systems.<br>• Challenge to reactor structures including confinements or containments.<br>• 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.<br>• Facilities in this category are insensitive to changes in the seismic hazard since they are air coolable in the event of a seismic induced loss-of-coolant accident.<br>• Building code bases is to prevent building collapse.<br>• Loss of air cooling from debris obstruction highly unlikely.<br>• Fuel cladding remains intact.<br>• No radiological release expected. |

### 3.3. Potential Impact of Beyond-Design-Basis External Events to Category 2 Research Reactors

In its assessment of the potential impact of beyond-design-basis external events to Category 2 RTRs, the NRC staff considered two research reactors operating at 6 $MW_t$ and 10 $MW_t$, respectively, and one 20 $MW_t$ test reactor.

The research reactors at the Massachusetts Institute of Technology (MITR) and the University of Missouri at Columbia (MURR) represent the two highest-powered research reactors in Category 2. These are tank-type reactors capable of adequately removing decay heat by the natural convection of the reactor coolant following a severe external event with a concurrent loss of all electrical power and active decay heat removal systems. Therefore, in this scenario, there would not be a near-term need to replenish the water around the reactor core lost through evaporation. However, when an initiating external event also results in a loss of primary coolant (i.e., a failure of the core tank and reactor pool integrity), these reactors would enter into inadequate decay heat removal conditions.
A seismic event could result in a loss-of-coolant accident (LOCA) concurrent with a loss of AC power. Such an event would need to significantly exceed the predictive potential for

[1] 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 would not have a detrimental effect on decay heat removal, which can be achieved by air cooling alone.

seismic activity at the sites of these reactors in order to cause the failure of the core tanks and reactor pools. For example, the MITR is located in an area where the U.S. Geological Survey's (USGS's) 2014 seismic hazard data predict a maximum peak ground acceleration (PGA) of 0.16 g [3]. A seismic analysis referenced in the MITR Safety Analysis Report demonstrates that the reactor core tank is capable of withstanding, at yield stresses, simultaneous static forces corresponding to 5.1 g horizontal and 3.4 g vertical [6]. Based on this analysis, the licensee concluded that the seismically-induced LOCA is not credible.

In 2006, a seismic assessment was conducted for the MURR containment building to assess its resistance to a seismic event [7]. The seismic response spectrum applied the Safe Shutdown Earthquake (SSE) used at the Callaway Nuclear Plant site, anchoring at a PGA of 0.2 g. The rationale for selecting the Callaway Nuclear Plant SSE was the plant's close proximity to the MURR.

For both the MITR and MURR, the NRC staff performed a probabilistic seismic hazard analysis (PSHA) to assess the safety of the facilities using present-day methodologies described in NRC Regulatory Guide (RG) 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," and in the 2012 Electric Power Research Institute (EPRI) report, "Seismic Evaluation Guidance, Screening, Prioritization and Implementation Details for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" [8, 9]. As inputs, the NRC staff used the Central Eastern United States Seismic Source Characterization (CEUS-SSC) model described in NUREG-2115, along with the updated 2013 EPRI ground motion model [10, 11]. As part of its assessment, the NRC staff included all CEUS-SSC background seismic sources within a 500 kilometer (km) [300 mile (mi)] radius and all repeated large magnitude earthquake (RLME) sources falling within a 1,000 km [620 mi] radius of the MITR and MURR sites. The NRC staff used the resulting base rock seismic hazard curves together with a confirmatory site response analysis to develop control point seismic hazard curves.

Based on its evaluation of the seismic hazard at the MITR site, the NRC staff determined that the estimated PGA values between 0.2 g and 0.225 g approximate the previously analyzed SSE of 0.225 g for the facility. For the MURR, the NRC staff determined that the ground motion response spectra is enveloped by the facility's previously analyzed SSE in the 1 to 16 hertz (Hz) range. For high frequency vibrations above 16 Hz, the electrical relays would disengage, dropping the control rods and shutting down the reactor. Moreover, due to the passive "fail safe" nature of the design, no electrical power is needed for safe shut down and maintaining cooling of the reactor. The NRC staff also found that sloshing induced by a seismic event would also not create any additional hazard at either facility. Therefore, based on its evaluations, the NRC staff concluded that no additional assessment was needed for seismic-related hazards at either the MITR or MURR [12, 13].

The NRC staff also assessed the capability of other beyond-design-basis external events (e.g., missiles from high winds) resulting in a LOCA concurrent with a loss of electrical power. Both the MITR and MURR have containments and designs that protect the core with an above-grade reinforced concrete biological shield that serves as an additional barrier to minimize the potential for impact damage to the reactors' pool liners, core tanks, and fuel.

The NRC staff has assessed the potential for damage of the MITR and MURR cores from a wind-related phenomenon using current tornado information given in NRC RG 1.76, "Design-Basis Tornadoes and Tornado Missiles in Nuclear Power Plants" [14]. The NRC staff selected a rigid large tornado missile, such as Schedule 40 pipe, striking the exterior walls of the containment building of each facility for its assessments. The NRC staff concluded that, upon striking, the pipe would be able to perforate the walls of these facilities. However, in doing so, the missile would lose a substantial amount, if not all, of its kinetic energy, making it unlikely to reach the biological shield of the MURR. For the MITR, such a missile would be unable to penetrate the steel plates in the domed portion of the containment structure. Thus, the biological shield surrounding the MITR would not experience any impact of a tornado missile strike. Therefore, the NRC staff concluded that it is unlikely that the MITR or MURR would experience any substantial damage from a rigid tornado missile strike, and no additional assessment is necessary for high-wind-related hazards at these facilities [12, 13].

### 3.4. Potential Impact of Beyond-Design-Basis External Events to Category 2 Test Reactor

The National Institute of Standards and Technology test reactor (NBSR), the third facility considered in Category 2, is a heavy water moderated and cooled tank-type reactor designed to operate at 20 $MW_t$. The reactor features a passive coolant makeup system and natural convection cooling that can protect against a fuel cladding failure following the loss of active decay heat removal systems. The passive coolant makeup system consists of a reserve tank within the reactor tank and an emergency tank external to the reactor tank. The inner reserve tank can make up the coolant inventory boiled off during the first half hour of an event. The emergency tank can then continue to replenish coolant for an additional 2 hours. Should the coolant in the passive makeup system become depleted during an event, the coolant in the reactor tank would continue to boil off, uncovering the reactor fuel. In such a scenario, fuel temperatures could rise to the point of cladding failure unless active decay heat removal systems are restored or otherwise provided via portable equipment. To restore active decay heat removal systems, emergency power can be supplied by either the 125-volt direct current station batteries or from one of the two onsite emergency diesel generators. Additionally, light water can be supplied to the reactor tank through the installation of a spool piece. In the event of a LOCA, heavy water primary coolant that is lost from the reactor tank can be collected in sumps and then pumped back into the reactor tank for decay heat removal; however, this action requires the availability of AC power. Based on these design features and limitations, the fuel cladding of the NBSR is vulnerable to failure should a beyond-design-basis seismic event result in an extended loss of electrical power, active decay heat removal systems, or coolant inventory makeup capability.

The NBSR is located in a low seismic activity zone. As described in its safety analysis report, the NBSR has been designed to withstand the stresses of an earthquake with a PGA of 0.1 g [15, 16]. Based on the USGS seismic hazard map, the probability of a seismic event occurring that would exceed this design basis earthquake is 2 percent in 50 years – corresponding to a return period of 2,475 years.

For the NBSR, the NRC staff performed a PSHA to assess the safety of the facilities using present-day methodologies described in NRC RG 1.208 and in the 2012 EPRI report addressing NTTF seismic recommendations. As inputs, the NRC staff used the CEUS-SSC model described in NUREG-2115, along with the updated 2013 EPRI ground motion model. As part of its assessment, the NRC staff included all CEUS-SSC background seismic sources within a 500 km [300 mi] radius and all RLME sources falling within a 1,000 km [620 mi] radius of the NBSR site. The NRC staff used the resulting base rock seismic hazard curves together with a confirmatory site response analysis to develop control point seismic hazard curves.

As part of its evaluation of the seismic hazard at the NBSR site, the NRC staff calculated the PGA for the NBSR site from the PSHA and site response analysis. For the NBSR, the NRC staff estimated the PGA value to be 0.07 g with a return period of 2,500 years. This estimated PGA compares well with the 0.1 PGA in the NBSR design basis earthquake. As such, the passive coolant makeup system, natural convection cooling, and availability of portable equipment provide adequate additional safeguards against radiological consequences during a seismic event. The NRC staff also found that sloshing induced by a seismic event would also not create any additional hazard at either facility. Therefore, based on its evaluations, the NRC staff concluded that no additional assessment was needed for seismic-related hazards at the NBSR [17].

There are no major bodies of water near the NBSR and the facility is located above the 500-year flood plain. As such, there are no water control structures, such as dams, dikes, or levees whose failure could result in the flooding of the facility. Based on the NRC staff's review of the watershed and location of the NBSR, the only flooding mechanisms for the NBSR site are rivers, streams, and local intense precipitation (LIP) events. As part of its evaluation of potential flooding events, the NRC staff confirmed that the NBSR site is outside of the 500-year flood zone using the FEMA flood insurance rate map. Therefore, the NRC staff concluded that river and stream flooding is not a credible hazard for the site [17]. And, while expected to be rare occurrences, flooding from LIP events, resulting in an extended loss of electrical power and active decay heat removal, would not adversely impact the availability of heavy and light water makeup sources for the core. Makeup water can be replenished through connections to the city water system, which do not require electrical power. Therefore, the NRC staff concluded that flooding would not have a significant impact on the NBSR site, and no additional assessment is necessary for flooding-related hazards at this facility [17].

The NRC staff also assessed the capability of other beyond-design-basis external events (e.g., missiles from high winds) resulting in a LOCA concurrent with a loss of electrical power. The NBSR is surrounded by a concrete biological shield located inside a confinement building. The exterior walls of the confinement building are made of 0.6-meter (m) [2 feet (ft)]-thick concrete.

The NRC staff has assessed the potential for damage of the NBSR core from a wind-related phenomenon using current tornado information given in NRC RG 1.76. The NRC staff selected a rigid large tornado missile, such as Schedule 40 pipe, striking the exterior walls of the confinement building for its assessments.[2] The NRC staff concluded that, upon striking the confinement building perpendicularly at a speed of 34 m/second (s) [112 ft/s], the pipe would be able to perforate the wall of the NBSR to a depth of 0.1 m [0.3 ft]. The estimated scabbing thickness, measured from the interior of the concrete wall, would be 0.5 m [1.7 ft]. However, because the exterior wall of the confinement building is 0.6 m [2 ft] thick, the hypothetical tornado missile would not reach the biological shield of the NBSR. Therefore, the NRC staff concluded that the NBSR would not experience any substantial damage from a

[2] For Gaithersburg, Maryland, the location of the NBSR, the expected speed of tornado missiles is larger than the expected speed of any hurricane-generated missiles at the same annual frequency of exceedance [18]. Therefore, the tornado missiles would be bounding in the damage assessment from wind-generated missiles.

rigid tornado missile strike, and no additional assessment is necessary for high-wind-related hazards at this facility [17].

## 4. Assessment of NTTF Recommendations and Conclusions

Based on its assessment of the NTTF recommendations, the NRC staff concluded that the research reactors in Category 1 and the two 2 MWt Category 2 research reactors are resilient to cladding failure due to decay-heat-induced overheating. This resiliency is attributable to the low power and infrequent operation of these facilities. These reactors do not rely on the availability of electrical power, active decay heat removal systems, or coolant to adequately remove decay heat. Even if a loss of coolant event results from or occurs concurrently with an external event, decay heat could be sufficiently removed via air cooling of the core. Confinement or containment structures are not challenged by energetic releases from primary or secondary cooling systems or by hydrogen generation from metal-water reactions during accidents. The research reactors in this category do not generate significant quantities of spent fuel and the minimal quantities that exist can be adequately cooled by air. Therefore, the Category 1 research reactors and two 2 $MW_t$ Category 2 research reactors present minimal radiological hazards to the public health and safety.

Based on its assessment of the NTTF recommendations, the NRC staff concluded that the three highest power Category 2 reactors (i.e., the MURR, MITR, and NBSR) are sensitive to the availability of reactor coolant. The loss of reactor coolant could result in failure of the fuel cladding and subsequent radiological release unless reactor coolant makeup is provided. If coolant remains available during an event, all three of these reactors would initially have adequate decay heat removal via natural circulation cooling. The two research reactors assessed in this category do not rely on active decay heat removal systems and electrical power to prevent fuel cladding failure. However, an extended loss of electrical power that prevents the operation of the active decay heat removal systems of the NBSR could result in a fuel cladding failure without coolant makeup from portable external sources.

Confinement or containment structures at these facilities are not challenged by energetic releases from primary or secondary cooling systems or by hydrogen generation from metal-water reactions during accidents. Additionally, the reactors in this category do not generate significant quantities of spent fuel and the minimal quantities that exist can be adequately cooled by air.

In its evaluation of seismic, flooding, and high-wind-related hazards, the NRC staff determined that the existing design bases for these facilities adequately protect against cladding failures and the release of radioactive material during an external event that could result in the failure of the primary coolant system integrity. As such, these three facilities present minimal radiological hazards to the public health and safety. Therefore, no additional protective or mitigating strategies are necessary for the MURR, MITR, and NBSR facilities in Category 2.

## 5. References

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