December 16, 2016

Mr. Al Queirolo, Director of Reactor Operations
Massachusetts Institute of Technology
Nuclear Reactor Laboratory
Research Reactor
138 Albany Street, MS NW12-116A
Cambridge, MA 02139

SUBJECT: MASSACHUSETTS INSTITUTE OF TECHNOLOGY – STAFF ASSESSMENT OF APPLICABILITY OF FUKUSHIMA LESSONS LEARNED TO THE MASSACHUSETTS INSTITUTE OF TECHNOLOGY RESEARCH REACTOR

Dear Mr. Queirolo:

The purpose of this letter is to provide you with the results of the U.S. Nuclear Regulatory Commission (NRC) staff's assessment of the applicability of Fukushima lessons learned to the Massachusetts Institute of Technology Research Reactor (MITR). In a letter dated May 27, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15112A126), you were informed of the NRC staff's intention to perform an audit of the MITR to determine if additional regulatory action at your facility was necessary based on Fukushima lessons learned.

The NRC staff performed a preliminary assessment for research and test reactors that is documented in "Draft White Paper Applicability of Fukushima Lessons Learned to Facilities other than Operating Power Reactors," (ADAMS Accession No. ML15042A367) dated March 2, 2015. The assessment was further updated, finalized, and provided to the Commission in SECY 15-0081, "Staff Evaluation of Applicability of Lessons Learned from the Fukushima Dai-Ichi Accident to Facilities Other Than Operating Power Reactors," (ADAMS Accession No. ML15050A066). These assessments identified the need for the NRC staff to perform additional evaluations for the MITR. The May 27, 2015, audit plan describes the scope of the NRC staff's information needs to support these additional evaluations to determine whether or not additional regulatory action is needed for the MITR based on Fukushima lessons learned.

The enclosure to this document provides the results of the NRC staff's assessment of your facility. The assessment is based on information provided during the audit as well as information that is available on the MITR Docket No 50-20. The NRC staff assessment concludes that current regulatory requirements for the MITR serve as a basis for reasonable assurance of adequate protection of public health and safety and that no additional regulatory actions are necessary.

Should you have any questions concerning this audit, please contact Mr. Patrick Boyle at (301) 415-3936 or by electronic mail at Patrick.Boyle@nrc.gov.

Sincerely,

/RA/

Alexander Adams, Jr., Chief
Research and Test Reactors Licensing Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-020
License No. R-37

Enclosure:
As stated

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OFFICE	NRR/DPR/PRLB	NRR/DPR/PRLB/LA*	NRR/JLD/JPSB/BC	NRR/DPR	NRR/DPR/PRLB/PM
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OFFICIAL AGENCY RECORD

STAFF ASSESSMENT OF APPLICABILITY OF FUKUSHIMA LESSONS LEARNED TO THE MASSACHUSETTS INSTITUTE OF TECHNOLOGY RESEARCH REACTOR LICENSE NO. R-37; DOCKET NO. 50-20

1.0 INTRODUCTION

The NRC staff identified the need for additional information for three high-power research and test reactors (RTRs) (including the Massachusetts Institute of Technology research reactor (MITR)) in a preliminary assessment dated March 2, 2015, "Draft White Paper Applicability of Fukushima Lessons Learned to Facilities other than Operating Power Reactors." The draft white paper can be found in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML15042A367. The assessment was further updated, finalized, and provided to the Commission in SECY 15-0081, "Staff Evaluation of Applicability of Lessons Learned from the Fukushima Dai-Ichi Accident to Facilities Other Than Operating Power Reactors" (ADAMS Accession No. ML15050A066).

As discussed in SECY 15-0081, the MITR is a tank type non-power reactor licensed to operate at a maximum thermal power level of 6 megawatts (MWt). Because of the MITR's power level, the designed natural convection flow of reactor coolant is sufficient to remove decay heat from the reactor and prevent bulk boiling, even in the event of a loss of all electrical power and active decay heat removal systems. Therefore, there is not a near-term need to replenish the water around the reactor fuel lost by evaporation. This assessment documents the NRC staff's review of the licensee's ability to address scenarios involving extreme external events which could potentially result in a loss of coolant inventory sufficient to cause inadequate decay heat removal and possible fuel damage. As stated in SECY 15-0081, NRC staff is performing these additional assessments related to seismic and high-wind events potentially resulting in a failure of primary coolant integrity.

2.0 REGULATORY EVALUATION

The purpose of the NRC staff's evaluation was to determine if additional regulatory action was necessary for the MITR based on Fukushima lessons learned. SECY 15-0081 Enclosure 1, Section 8 provides a background regarding licensing of RTRs. The discussion found in this section of SECY 15-0081 includes the following background:

> The NRC's authority to license and regulate non-power reactors (NPRs) is provided in Sections 103 and 104 of the Atomic Energy Act (Act) as amended. Section 103 of the Act pertains to the licensing of industrial or commercial reactors that can consist of both power and NPRs. Section 104 of the Act relates to the licensing of NPRs for the purpose of medical therapy and research and development. All RTRs currently licensed by the NRC are licensed under Section 104 of the Act. Unique to this authority are the provisions contained in Paragraph 104c of the Act that directs the "Commission to impose the minimum amount of such regulation and terms of license that will permit the

> Commission to fulfill its obligation under this Act to promote the common defense and security and to protect the health and safety of the public with the intent to permit the conduct of widespread and diverse research and development."

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

As part of its assessment, the NRC staff considered the provisions of Section 104c of the Act and whether additional regulatory actions were necessary based on Fukushima lessons learned.

3.0 TECHNICAL EVALUATION

3.1 Applicability of Fukushima Lessons Learned to the MITR

SECY 15-0081 provides a detailed evaluation of the applicability of Fukushima lessons learned to the MITR. Specifically, SECY 15-0081, Enclosure 1, Section 8 states that the MITR is a tank type reactor capable of removing adequate decay heat by the natural convection flow of the reactor coolant following a severe external event even if that event results in the loss of all electrical power and active decay heat removal systems. In this case, decay heat is not sufficient (given the passive heat sink) to raise the temperature of the primary coolant above bulk boiling. Therefore, for this scenario, there is no near-term need to replenish the primary coolant 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 integrity, do the conditions exist that result in inadequate decay heat removal.

The radiological consequences resulting from a severe external event may exceed those assumed in a maximum hypothetical accident (MHA)[1] but would not exceed Title 10 of the *Code*

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

of Federal Regulations (10 CFR) Part 100, "Reactor Site Criteria." Although 10 CFR Part 100 dose criteria[2] are not applicable to the MITR, the criteria was used as part of a post-9/11 security assessment. This security assessment of sabotage scenarios assumed massive damage states to the facility. Because of the malicious intent and the extreme assumptions of facility damage used in the sabotage assessment, the postulated radiological consequences from the worst case sabotage event are expected to bound the postulated radiological consequences of all external events. The radiological consequences predicted by the worst case sabotage event analysis are a fraction of the 10 CFR Part 100 reactor siting dose criteria. The security assessment discussion is provided for reference because it was considered in the NRC staff assessment as to whether additional regulatory actions for the MITR are needed.

As stated in SECY 15-0081, the NRC staff assessed beyond design-basis events, such as missiles created by high winds and seismic events, to determine if additional regulatory actions are needed to address these events.

3.2 NRC Staff Assessment of Potential for a Beyond Design Basis Natural Phenomena Event to Cause Core Damage at the MITR

The NRC staff has assessed the seismic and high wind-related hazards using the latest information and guidelines.

3.2.1 Brief Description of the Containment Building and the Reactor

The MIT reactor is heavy water reflected and light water cooled and moderated research reactor (MIT, 1999). The reactor is located in the center of the main floor of the containment building with a 7.6 m [25 ft] wide annular ring of floor space surrounding it. The containment building is a domed cylindrical structure. It has a diameter of 22.5 m [74 ft] and an above-grade height of 14.9 m [49 ft]. The cylindrical portion of the structure is made of 9.5 mm [3/8 in]-thick ASTM 283 Grade C steel plate surrounding a 0.6 m [2 ft]-thick concrete wall. The concrete wall is 9.6 m [31.5 ft] high. The dome portion is made of 15.8 mm [5/8 in] thick steel plate. The plates are welded to form an airtight structure (MIT, 1999). The basement is made of reinforced concrete. In addition to the control room, several other facilities are located in the basement (MIT, 1999).

The reactor has dual, concentric tanks. The outer tank contains heavy water and the inner tank contains light water. The reactor core is located within the light water tank (inner tank). This tank is made of high purity aluminum and is a closed system. The light water tank is supported by the heavy water tank. The outer tank has a diameter of 1.2 m [4 ft]. The light water in the inner tank serves as both the primary coolant and moderator. Approximately 0.6 m [2 ft] wide graphite reflector surrounds the outer tank (MIT, 1999). The reactor tanks have both a thermal shield and a biological shield surrounding them. The tanks are attached to an annular ring, which, in turn, is bolted to the inside steel cylinder of the thermal shield (MIT, 1999). The radial section of the thermal shield consists of an inner steel cylinder of 2.4 m [8 ft] in diameter and

[2] The 10 CFR Part 100 dose criteria are as follows: 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.

5 cm [2 in] in thickness, a 3.8 cm [1.5 in] thick lead-filled space, and an outer steel cylinder with 5 cm [2 in] thickness (MIT, 1999). The biological shield is made of dense concrete having a thickness of approximately 1.7 m [5.5 ft].

3.2.2 Seismic Assessment Basis of the MITR Facility

A seismic event can result in a loss of coolant accident (LOCA) and concurrent loss of ac power to the MITR facility. The MITR facility is in an area where the U.S. Geological Survey (USGS) 2014 seismic hazard data predict a maximum peak ground acceleration (PGA) of 0.16g (NRC, 2015). The safety analysis report (SAR) (MIT, 1999) has considered the Safe Shutdown Earthquake (SSE) for the MITR facility to be 0.225g, assuming an amplification of 1.5 times of the bedrock acceleration of 0.15g in the overlain soil layers by referring the Pilgrim Nuclear Power Station, located 96 km [60 mi] southeast (MIT, 1999). An analysis referred in the SAR (MIT, 1999) shows that there would be no damage to the core tank below a combined 5g horizontal and 3.4g vertical accelerations because the peak stress estimated on the aluminum tank from the ground motion would only reach the yield stress of aluminum.

3.2.2.1 Staff Confirmatory Assessment

RTRs are small in power output capacity and, consequently, their seismic designs generally use less site-specific information and are less stringent than a commercial power reactor. Their SAR either presents only a PGA or refers to a nearby commercial reactor's SSE as their design ground motion, as done for the MITR facility.

The NRC staff has assessed whether the reevaluated seismic hazard at the MITR facility site creates a significantly higher seismic hazard. The NRC staff has performed a probabilistic seismic hazard analysis (PSHA) for the MITR site to assess the seismic safety of the MITR facility using present-day methodologies, as described in Electric Power Research Institute guidance (EPRI, 2012), and Regulatory Guide (RG) 1.208 (NRC, 2007a). As an input, the NRC staff used the Central Eastern United States Seismic Source Characterization (CEUS-SSC) model in NUREG–2115 (NRC, 2012) along with the updated EPRI ground motion model (EPRI, 2013). Consistent with EPRI guidance (EPRI, 2012), the NRC staff included all CEUS-SSC background seismic sources within a 500 km [310 mi] radius of the MITR site. In addition, the NRC staff included all of the repeated large magnitude earthquake (RLME) sources falling within a 1,000 km [620 mi] radius of the site. For each of the CEUS-SSC sources used in the PSHA, the NRC staff used the mid-continent version of the updated EPRI ground motion model (EPRI, 2013). 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 and estimate the PGA for comparison with the SSE.

The control point is not specified in the SAR (MIT, 1999). The NRC staff assumed for this assessment that the control point is located at the top of the surface following the suggestions given in EPRI guidance (EPRI, 2012). The current surface after removal of the fill and organic layers during construction of the reactor is taken as the horizon of the control point. The seismic motion would be estimated and compared at the control point location.

The purpose of the site response analysis is to determine the site amplification that will occur as a result of bedrock ground motion propagating upwards through the soil/rock column to the

surface. The critical parameters that determine what frequencies of ground motion are affected by the upward propagation of bedrock motions are the layering of soil and/or soft rock, the thicknesses of these layers, the shear-wave velocities and low-strain damping of the layers, and the degree to which the shear modulus and damping change with increasing input bedrock motion amplitude.

To estimate the parameters necessary to conduct the site response analysis, the NRC staff studied available information from the MITR facility, nearby Pilgrim Nuclear Power Station, and other relevant studies conducted in the city of Boston and surrounding areas. The bedrock at the site is Cambridge argillite, as is in most of Boston area (MIT, 1999). It is typically hard and thinly laminated to massive; however, it could be severely weathered to kaolinite (Hayes, et al., 2001). The artificial fill and organic layers at the top were removed during construction of the reactor facility (MIT, 1999). Currently, the top sand lay is underlaid by a thin silt layer and a very thick clay layer, called Boston Blue Clay, over the bedrock (MIT, 1999). The Boston Blue Clay is a marine clay unit resulted from the advance of the sea immediately following the retreat of the glacial ice (Hayes, et al., 2001). Due to glacial erosion, the top of bedrock (Cambridge argillite) is highly variable. The SAR (MIT, 1999) does not provide the depth of bedrock at the site. However, presence of Boston Blue Clay is often an indicator of deep soil sites (Baise, et al., 2016). The stratigraphy at the Pilgrim Nuclear Power Station is quite different. Consequently, it was not possible to infer the depth of bedrock at the MITR site from information at the Pilgrim Nuclear Power Station site. Based on available information (e.g., Baise, et al., 2016), the depth of Cambridge argillite varies between 5 and 50 m [15 to 150 ft], but can be as high as 80 m [250 ft].

Additionally, shear wave velocity of the underlying subsurface materials was not reported in the SAR (MIT, 1999). The NRC staff has conducted an extensive literature search for information available from the surrounding areas. USGS (Thompson, et al., 2014) measured the subsurface shear wave velocity at 27 locations around the City of Boston. Measurement Station 909BUB is nearest to the MITR site, although outside the campus. The NRC staff has used the shear wave velocities of sand and silt layers, as measured in Station 909BUB, in the assessment. Measurement at Station 909BUB ended in Boston Blue Clay at a depth of 50 m [160 ft] but never reached the bedrock horizon. Trudeau, et al. (1974) made four borings at the west end of the campus. Weston Geophysics measured the in situ shear wave velocity in Boston Blue Clay. The measured shear wave velocity in different tests varied between 240 to 270 m/s [800 to 890 ft/s]. The NRC staff has selected the lowest measured value of 240 m/s [800 ft/s] to be conservative. When compared to the measured value at Station 909BUB, the selected value is smaller, the difference increases with depth. USGS Stations 912SFR and 926WAT measured the shear wave velocity of Cambridge argillite as at least 1,300 m/s [4,300 ft/s] at a depth of approximately 40 m [130 ft]. These stations are near the MITR facility (Thompson, et al., 2014).

Based upon the available information, there are uncertainties on the depth of the bedrock at the MITR site. In addition, it is reasonable to expect that there will be a weathered zone with substantially decreased shear wave velocity at the top region of the bedrock due to glacial erosion, as reported by Hayes, et al. (2001); however, its thickness at the MITR site is unknown. The NRC staff has conducted a sensitivity study with two values of the parameters to assess how the estimated site amplification is affected by varying the weathered zone thickness.

Two values, 7.5 m [25 ft] and 15 m [50 ft], are used as thickness of the weathered zone with shear wave velocity assumed as 303 m/s [1,000 ft/s]. In both cases (MIT10 and MIT6), the shear wave velocity abruptly rises to 2,800 m/s [9,200 ft/s] as the bedrock is encountered. Bedrock is defined for site response analysis as the layer with shear wave velocity of at least 2,800 m/s [9,200 ft/s]. Consequently, the NRC staff has studied two other cases where the shear wave velocity gradually increases to 2,800 m/s [9,200 ft/s] at a depth of 160 m [535 ft] from the surface. These cases (MITR1 and MITR2) are similar to the observation at the USGS Station 912SFR (Thompson, et al., 2014). Based upon the qualitative information available from the SAR (MIT, 1999) and literature, all four are probable profiles to represent the subsurface materials at the site. Lack of site-specific information precludes selecting one profile as the preferred one or assigning weights to these profiles. Consequently, the NRC staff has used all four profiles in the assessment. These shear wave velocity profiles are shown in Figure 3.2-1.

The MITR SAR (MIT, 1999) did not report any dynamic material properties. Literature search was conducted for information on the shear modulus degradation and associated damping of the subsurface materials. Chakraborty et al. (2013) have reported the shear modulus degradation curve for Boston Blue Clay; but they did not provide the associated damping curve. The shear modulus degradation curve has been digitized and was used in this assessment along with (EPRI, 1993) soil damping curve with Plasticity Index (PI) of 30. Following Baise et al. (2016), the sand layer has been modeled using Seed and Idriss (1970) sand curve; the silt layer has been represented by Vucetic and Dorby (1991) clay curve with PI equal to 30, and the EPRI rock curve has been used to model the weathered zone and the transition zone used in some simulations. The bedrock has been modeled as linear and no damping rock layer.

Kappa is measured in units of seconds and is the damping contributed by both intrinsic hysteretic damping as well as scattering due to wave propagation in heterogeneous material. Mean base-case estimate of the kappa has been developed following EPRI, 2013. Kappa for a CEUS rock site with at least 1,000 m [3,000 ft] of sedimentary rock may be estimated from the average shear wave velocity over the upper 30 m [100 ft] of the subsurface profile with an additional kappa of 0.006 s for the underlying hard rock. The estimated base-case kappa is 0.0127 s. Epistemic uncertainty of kappa has been captured by two additional kappa values of 0.0081 s and 0.0199 s.

The aleatory variability of the dynamic material properties (namely, G/G_{max} and hysteretic damping curves) is also considered. Consequently, variability of the shear wave velocity profiles are developed from the base case profiles. Parameters developed by Toro (1997) for USGS “A+B” site conditions are used to model the correlation between layering and shear wave velocity. The random velocity profiles are generated using a natural log standard deviation of 0.35. The NRC staff used the random vibration theory approach to perform the site response analyses.

The NRC staff has estimated the seismic hazard curves at the control point; i.e., at the surface. The results are shown in Figure 3.2-2. Figure 3.2-3 shows the median amplification function at a PGA of 0.2g for the four different profiles used. Differences among the four profiles used are more prominent in the frequency range of approximately 0.5 and 5 Hz.

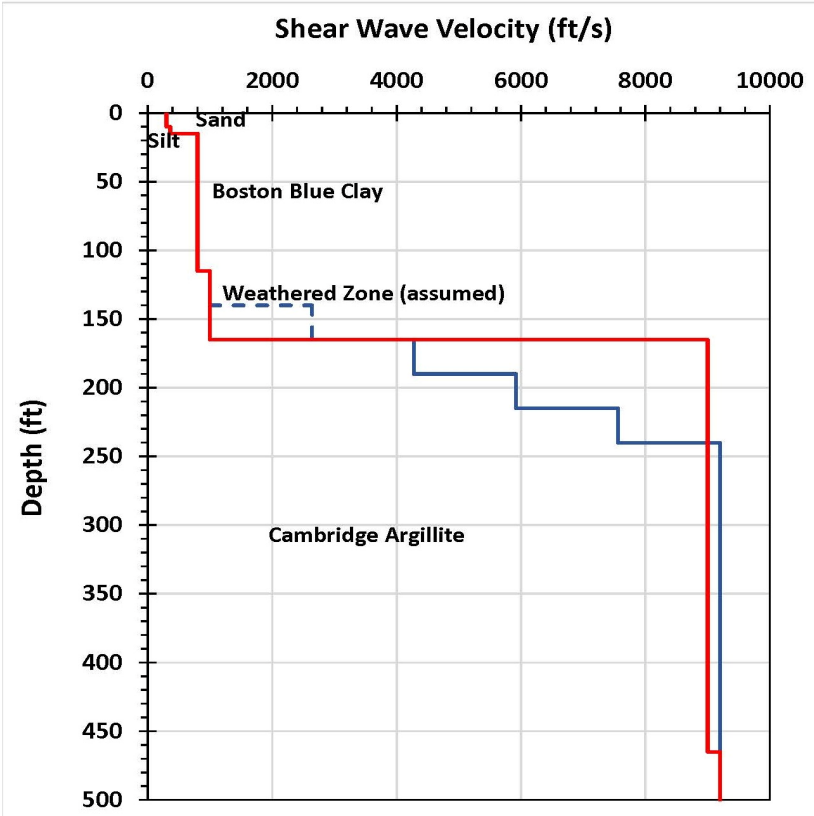

Figure 3.2-1. Shear Wave Profile for the MIT Reactor Site. Two Alternative Scenarios Assumed for Shear Wave Velocity Variation in Cambridge Argillite Are Shown as Dashed Lines.

The estimated values of PGA at the MITR site are between 0.2g and 0.225g, approximately the SSE of 0.225g for the facility. In addition, the reactor has a passive design and can remove decay heat by natural convective flow of the coolant following a severe external event even if the event results in loss of all electrical power and active decay heat removal systems (NRC, 2015). Therefore, based on this discussion, the NRC staff concludes that no additional assessment is needed for the MIT test reactor facility.

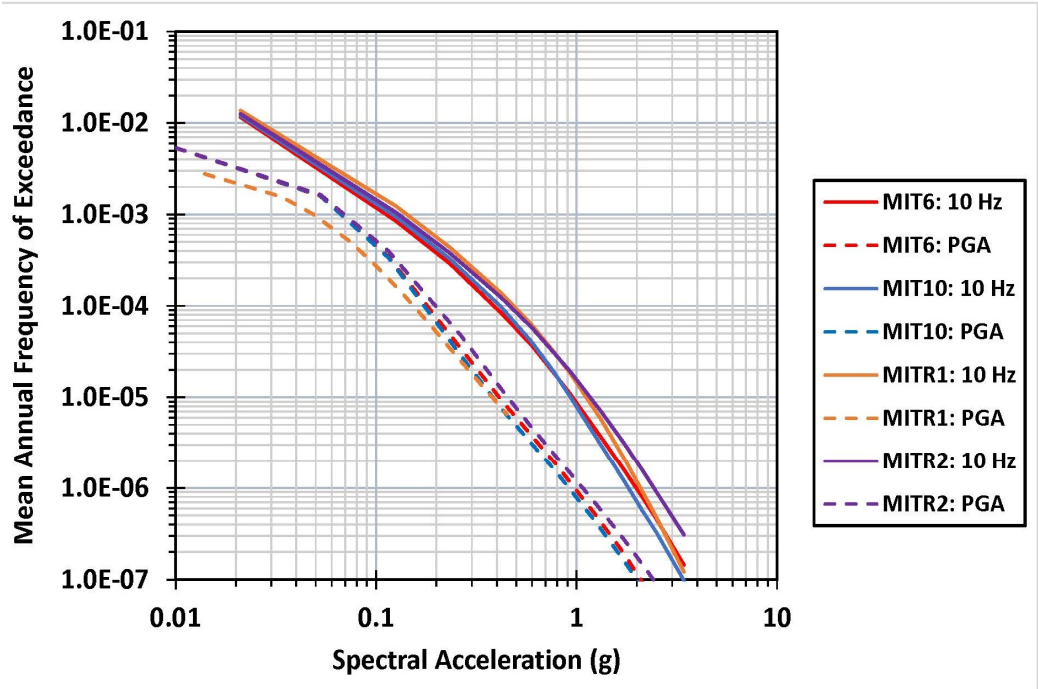

Figure 3.2-2. Mean Hazard Curves at Two Frequencies at the Control Point.

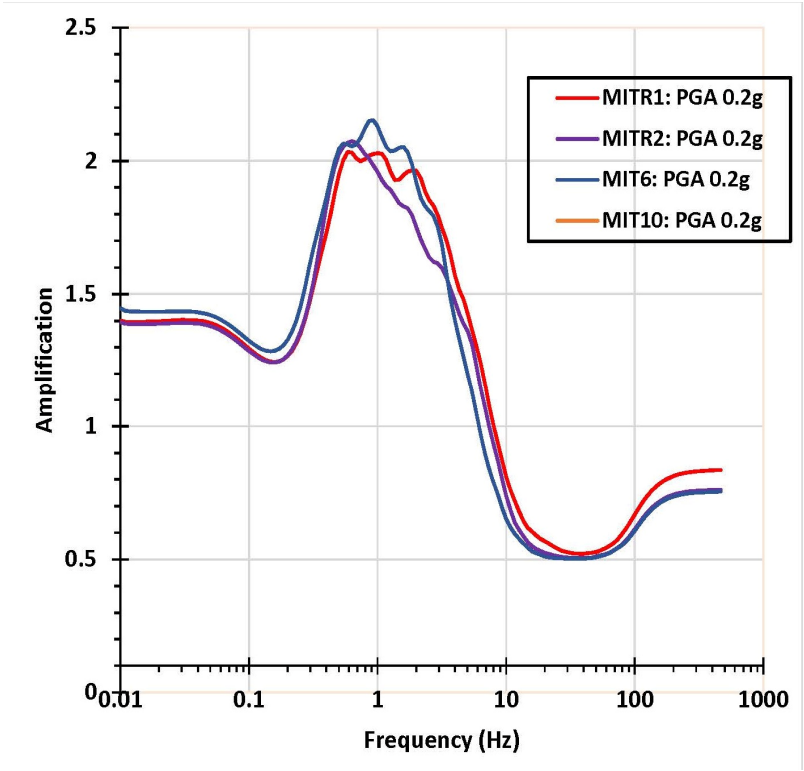

Figure 3.2-3. Median Amplification Function at a PGA of 0.2g

3.2.2.3 Effects of Sloshing from Seismic Event

The height of sloshing wave δ_s due to a seismic event with PGA of 0.225g has been estimated using both ASCE 7–10 (ASCE, 2013) and EPRI (EPRI, 2012) methodologies. The inside diameter of the inner tank is 1.12 m [44 in]. The height of water in the reactor tank is maintained at 3 m [10 ft]. Because the tank is anchored to the biological shield at the side, only overtopping of water due to sloshing induced by the seismic event is a potential concern. Damage from buckling of the tank wall or damage of the wall from sloshing-induced forces is not a concern. Similarly, sliding of the tank or base uplift are also not potential concerns.

Based on Table 20.3–1 of ASCE 7–10 (ASCE, 2013), the MITR site can be classified as Site Class D (average shear wave velocity in top 30 m [100 ft] is between 180 and 360 m/s [600 and 1200 ft/s]). Using Equation (15.7–12) of ASCE 7–10 (ASCE, 2013), the natural period of first convective mode of sloshing, T_c is estimated to be 1.11 s. The spectral acceleration of the sloshing water with $T_c = 1.11$ s and 0.5% damping is estimated to be 0.149 g using Equation (15.7–10) of ASCE 7–10 (ASCE, 2013).

The MITR facility may be categorized as a Risk Category IV structure following Table 1.5- 1 of ASCE 7–10 (ASCE, 2013) meaning it is an essential facility. Failure of this structure poses a substantial hazard to the community. It contains hazardous materials that can pose a threat to the public if released. Therefore, the Importance Factor, I_e, would be 1.5 according to Table 1.5–2 of ASCE 7–10 (ASCE, 2013). The estimated sloshing height, δ_s, using Equation (15.7–13) of ASCE 7–10 (ASCE, 2013) would be 10.5 cm [4.1 in].

Using Equation 7-1 of EPRI guidance (EPRI, 2012), the natural frequency, f_{c1}, for the fundamental convective (sloshing) mode of vertical oscillation of the water surface in a pool is estimated to be 0.84 Hz or, alternatively, the period is 1.2 s. The circular pool has been approximated by a square pool with each side equal to the diameter of the circular pool. Using Equation 7-2, the sloshing height is estimated to be 8.3 cm [3.3 in]. Following the suggestion given in EPRI guidance (EPRI, 2012), the estimated sloshing height is increased by 20% using Equation 7-3 to account for higher convective modes of sloshing and nonlinear sloshing effects. The revised sloshing height is 10 cm, practically the same as estimated using the ASCE 7–10 (ASCE, 2013) methodology.

It should be noted that the estimated sloshing height using either the ASCE 7–10 (ASCE, 2012) or the EPRI guidance (EPRI, 2012) methodology is appropriate for tanks or vessels storing liquids and supported only at the base. The top of the tank is free to oscillate. However, although the core at the MITR contains both heavy and light water within two concentric tanks, the top of these tanks are not free to oscillate. The heavy water tank is surrounded by approximately 0.6 m [2 ft]-thick graphite reflector. A 3.8 cm [1.5 in]-thick lead surrounds the graphite reflector and acts as the thermal shield of the reactor. The biological shield, made of high-density concrete, surrounds the thermal shield (MIT, 1999, Figures 4-1 and 4-10). Therefore, water in the core tank is extremely unlikely to develop such sloshing wave height as estimated using either of the two methods.

Nevertheless, the NRC staff compared the estimated wave height with the height of freeboard available, which is 7.5 cm [3 in]. The water level in the core tank is controlled by a 5 cm [2 in] diameter overflow pipe that maintains the coolant (light water) level at 7.5 cm [3 in] below the

top of the tank (MIT, 1999). Because of this design feature, any sloshing water above 7.5 cm [3 in] will flow out of the tank instead of overtopping it. Therefore, possibility of loss of light water coolant due to sloshing is very small. Additionally, the top shield lid of the reactor core would be in place closing the reactor tank if the reactor power level exceeds 100 kW. This additional operating feature would further reduce the potential for overtopping of coolant due to sloshing.

3.2.3 Assessment of Tornado-Generated Missile Strikes

The NRC staff assessed the potential of damage to the reactor core from a wind-related phenomenon at the MITR facility site using the current tornado information given in RG 1.76 (NRC, 2007b) and hurricane information given in Vickery, et al. (2011). The NRC staff also referred to Department of Energy (DOE) Standard DOE–STD–3014–2006 (DOE, 2006) to assess the response of a steel after a tornado-generated missile strike. The missile is assumed rigid in this analysis for maximum penetration. Only missiles generated by a tornado or a hurricane may have the potential to damage the core after striking the containment building. It should be noted that at Cambridge, Massachusetts, the location of the MIT reactor facility, the expected speed of the tornado missiles are larger than the expected speed of any hurricane-generated missiles at same annual frequency of exceedance (Vickery, et al., 2011). Therefore, the tornado missiles would be bounding in damage assessment from wind-generated missiles.

The NRC staff selected a rigid large tornado missile, such as, a Schedule 40 pipe, striking the exterior walls of the confinement building for this assessment because other missiles in the spectrum of missiles suggested in RG 1.76 (NRC, 2007b) would either deform on impact or require an opening in the protective barrier to pass through. Following RG 1.76 (NRC, 2007b), the missile is cylindrical in shape with diameter 0.168 m [6.625 in], weighing 130 kg [227 lb], and traveling at a speed of 34 m/s [112 ft/s]. Because Cambridge, Massachusetts, is located in Tornado Region II, a speed of 34 m/s [112 ft/s] is appropriate for the design-basis tornado having a wind speed with an annual exceedance frequency of 10^{-7} for a nuclear power reactor (NRC, 2007b).

The method given by DOE (2006) has been used to estimate the maximum depth of perforation of a steel plate when a Schedule 40 pipe strikes. The Schedule 40 pipe is assumed to strike horizontally the cylindrical portion of the containment structure at a speed of 34 m/s [112 ft/s]. This impact would perforate a steel plate to a depth of 8.7 mm [0.34 in] only. For the dome-shaped portion of the containment structure, the strike speed is taken as the resultant of the maximum horizontal speed of 34 m/s [112 ft/s] and two-third of this speed as the vertical speed, following RG 1.76 (NRC, 2007b). Using DOE (2006), the maximum perforation depth of a steel plate from a strike at 41 m/s [135 ft/s] would be 10.7 mm [0.42 in]. Because the steel plates in the domed portion of the containment structure have a thickness of 15.8 mm [5/8 in], the missile would not be able to perforate them. Similarly, the thickness of the steel plates is higher than the maximum perforation depth in the cylindrical portion of the containment structure. Additionally, there is a 0.61 m [2 ft] thick concrete wall concentric with the cylindrical portion of the containment structure. Therefore, the biological shield surrounding the reactor core would not experience any impact of a tornado missile strike, as the rigid missile would not be able to reach the shield. The tornado missile speed assumed here is for extremely rare tornados having a wind speed with an exceedance frequency of 10^{-7} per year, appropriate for an

operating nuclear power plant (NRC, 2007b). The MITR being a research reactor with thermal output of 6 MWt, the appropriate tornado for hazard assessment would be a more frequent one; (i.e., a tornado with higher annual frequency of occurrence). This tornado would generate a significantly lower wind speed and the impact speed of the tornado-generated missiles would be smaller.

Effects of a crushable missile, such as an automobile, will be much less severe, as most of the kinetic energy will be absorbed by the building exterior wall. Therefore, based on the above discussion, the NRC staff concludes with reasonable assurance that it is unlikely that the reactor at the MITR facility will experience any substantial damage from a rigid tornado missile strike even from an extremely rare one.

4.0 CONCLUSION

The NRC staff assessed the seismic hazard of the MITR facility using present-day methodologies (EPRI, 2012) and regulatory guidance. Based on the re-evaluated seismic hazard, the NRC staff concludes that no additional assessment would be needed for seismic hazards. Sloshing induced by the seismic event will also not create any additional hazard. Additionally, the containment structure would be able to withstand tornado-generated missiles based on current information and guidance.

The NRC staff assessment concludes that current regulatory requirements for the MITR serve as a basis for reasonable assurance of adequate protection of public health and safety and that no additional regulatory actions are necessary.

5.0 REFERENCES

Note: ADAMS Accession Nos. refer to documents available through NRC's Agencywide Documents Access and Management System (ADAMS). Publicly available ADAMS documents may be accessed through http://www.nrc.gov/reading-rm/adams.html.

ASCE. "Minimum Design Loads for Buildings and Other Structures." ASCE/SEI 7–10. Reston, VA: American Society of Civil Engineers. 2013.

Baise, L.G., J. Kaklamanos, B.M. Berry, and E.M Thompson. "Soil amplification with a strong impedance contrast: Boston, Massachusetts." Engineering Geology. 202. pp. 1–13. 2016.

Chakraborty, T., R. Salgado, and D. Loukidis. "A Two-Surface Plasticity Model for Clay." Computers and Geotechnics. 49. 170–190. 2013.

DOE. "Accident Analysis for Aircraft Crash into Hazardous Facilities." DOE Standard DOE-STD–3014–2006. Washington, DC: U.S. Department of Energy. 2006.

Electric Power Research Institute (EPRI). "Seismic Evaluation Guidance, Screening, Prioritization and Implementation Details [SPID] for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic." EPRI Report 1025287. November 27, 2012, ADAMS Accession No. ML12333A170.

Electric Power Research Institute (EPRI). "EPRI Ground Motion Model Review Final Report." June 3, 2013. ADAMS Accession No. ML13155A553.

Hayes, K.E., J.E. Ebel, and A. Urzua. "Microtremor Measurements to Obtain Resonant Frequencies & Ground Shaking Amplification for Soil Sites in Boston." Civil Engineering Practice, Journal of the Boston Society of Civil Engineers Section. American Society of Civil Engineers. Fall/Winter. 17–36. 2001.

MIT. "Safety Analysis Report for the MIT Research Reactor (MITR-III)." Rev. 1. Nuclear Reactor Laboratory. Cambridge, MA: Massachusetts Institute of Technology (MIT). 1999.

NRC. "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion." Regulatory Guide 1.208. Washington, DC: U.S. Nuclear Regulatory Commission. 2007a. ADAMS Accession No. ML070310619.

NRC. "Design-Basis Tornadoes and Tornado Missiles in Nuclear Power Plants." Regulatory Guide 1.76. Revision 1. Washington, DC: U.S. Nuclear Regulatory Commission. 2007b. ADAMS Accession No. ML070360253.

NRC. "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities." NUREG-2115, ADAMS stores the NUREG as multiple ADAMS documents, which are accessed through the web page http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2115/. Washington, DC: U.S. Nuclear Regulatory Commission. 2012.

NRC. "Staff Evaluation of Applicability of Lessons Learned from the Fukushima Dai-Ichi Accident to Facilities Other than Operating Power Reactors." SECY–15–0081. Washington, DC: U.S. Nuclear Regulatory Commission. 2015. ADAMS Accession No. ML15050A066.

Seed, H.B. and I.M. Idriss. "Shear Moduli and Damping Factors for Dynamic Response Analyses." EERC 70–10. Earthquake Engineering Research Center. Berkley , CA: University of California. December 1970.

Thompson, E.M., B. Carkin, L.G. Baise, and R.E. Kayen. "Surface Wave Site Characterization at 27 Locations Near Boston, Massachusetts, Including 2 Strong-Motion Stations." USGS Open File Report 2014–1232. Reston, VA: United States Geological Survey. 2014.

Toro, G.R. "Probabilistic Models of Site Velocity Profiles for Generic and Site-Specific Ground-Motion Amplification Studies." Appendix to Silva, W., N. Abrahamson, G. Toro, and C. Costantino. Description and Validation of the Stochastic Ground Motion Model. PE&A 94PJ20. Report to Brookhaven National Laboratory. El Cerrito, CA: Pacific Engineering and Analysis. 1996.

Trudeau, P.J., R.V. Whitman, and J.T. Christian. "Shear Wave Velocity and Modulus of a Marine Clay." Boston Society of Civil Engineers. 61(1). January, pp. 12–25. 1974.

Vickery, P.J., D. Wadhera, and L.A. Twisdale. “Technical Basis for Regulatory Guidance on Design-Basis Hurricane Wind Speeds for Nuclear Power Plants.” NUREG/CR–7005. Washington, DC: U.S. Nuclear Regulatory Commission. 2011. ADAMS Accession No. ML11335A031.

Vucetic, M., and R. Dobry, R. “Effect of Soil Plasticity on Cyclic Response.” Journal of Geotechnical Engineering. 117(1), 89–107. 1991.