December 8, 2016

Mr. Ralph Butler, Director
Research Reactor Center
University of Missouri-Columbia
Research Park
Columbia, MO 65211

SUBJECT: UNIVERSITY OF MISSOURI AT COLUMBIA – STAFF ASSESSMENT OF APPLICABILITY OF FUKUSHIMA LESSONS LEARNED TO UNIVERSITY OF MISSOURI - COLUMBIA RESEARCH REACTOR

Dear Mr. Butler:

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 University of Missouri at Columbia Research Reactor (MURR). In a letter dated June 1, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15112A094), you were informed of the NRC staff's intention to perform an audit of the MURR 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 MURR. The June 1, 2015, audit plan describes the scope of the NRC staff's informational needs to support these additional evaluations to determine whether or not additional regulatory action is needed for the MURR 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 MURR Docket No. 50-186. The NRC staff assessment concludes that current regulatory requirements for the MURR 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. Geoffrey Wertz at (301) 415-0893 or by electronic mail at Geoffrey.Wertz@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. 50-186
License No. R-103

Enclosure:
As stated

cc: See next page

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OFFICIAL AGENCY RECORD

STAFF ASSESSMENT OF APPLICABILITY OF

FUKUSHIMA LESSONS LEARNED TO THE UNIVERSITY OF

MISSOURI AT COLUMBIA RESEARCH REACTOR

LICENSE NO. R-103; DOCKET NO. 50-186

1.0 INTRODUCTION

The NRC staff identified the need for additional information for three high-power research and test reactors (RTRs) (including the University of Missouri at Columbia research reactor (MURR)) 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 MURR is a tank type non-power reactor licensed to operate at a maximum thermal power level of 10 megawatts (MWt). Because of the MURR's power level, the designed natural convection flow of the reactor coolant system 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 staff's review of the licensee's ability to address scenarios involving extreme external events which could potentially result in 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 missiles created by high winds 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 MURR 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 the 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 MURR

SECY 15-0081 provides a detailed evaluation of the applicability of Fukushima lessons learned to the MURR. Specifically, SECY 15-0081, Enclosure 1, Section 8 states that the MURR 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 and reactor pool 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 of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria." Although 10 CFR Part 100

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[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 fission product release 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 fission product release 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").

dose criteria[2] are not applicable to the MURR, 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 MURR 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 Staff Assessment of Potential for a Beyond Design Basis Natural Phenomena Event to Cause Core Damage at the MURR

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 MURR, along with its control room, is located inside a containment building in Columbia, Missouri. The exterior walls of the containment building are made of 0.3 m [1 ft]-thick reinforced concrete (University of Missouri, 2006). The inside dimensions of the containment building are approximately 20.4 m [67 ft] ×18.9 m [62 ft] × 19.7 m [64.5 ft]. The reactor containment building is surrounded by laboratories and support facilities extending one story above the grade level (University of Missouri, 2006, Figures 1.2, 1.6, and 1.7). The reactor core is at below-grade level. The reactor is located near the bottom of a cylindrically shaped, aluminum-lined pool approximately 3.0 m [10 ft] in diameter and 9.1 m [30 ft] in depth. The pool liner is surrounded by and anchored to a biological shield made of reinforced concrete. The biological shield is approximately 9.4 m [31 ft] tall. Its thickness varies from 0.9 m [3 ft] to 2.3 m [7.5 ft], with the smaller dimension at the top. The external surface of the biological shield is lined with 6.35 mm [0.25 in]-thick steel plate fastened by tie rods. The biological shield is supported by a 1.1 m [3.5 ft]-thick concrete pad poured directly onto a 3.7 m [12 ft] high caisson. It extends horizontally out 0.3 m [1 ft] beyond the biological shield in all directions and downward to a minimum depth of 0.15 m [6 in] below the bedrock at the lowest point around the edge of the caisson (University of Missouri, 2006). The uniaxial compressive strength of the reinforced concrete used to construct the biological shield is 21 MPa [3,000 psi] (University of Missouri, 2006).

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[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.

3.2.2 Seismic Assessment Basis of the MURR Facility

A seismic assessment was conducted for the MURR containment building to assess its resistance to a seismic event (University of Missouri, 2006). The seismic response spectrum used is the Safe Shutdown Earthquake (SSE) used at the Callaway Nuclear Plant site. The spectrum is essentially a Regulatory Guide (RG) 1.60 spectrum, anchoring at a peak ground acceleration (PGA) of 0.2g. It was used for assessing the structural adequacy of the shear walls and the containment in an SSE. The rationale for selecting the Callaway Nuclear Plant was that it is located approximately 48 km [30 mi] east of this facility (University of Missouri, 2006). The staff compared the SSE with the ground motion response spectra (GMRS) developed in the staff assessment.

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 design ground motion either uses a shorter return period or refers to a nearby commercial reactor's SSE, as has been done for the MURR facility.

The NRC staff performed a probabilistic seismic hazard analysis (PSHA) for the MURR site to assess the seismic safety of the MURR facility using present-day methodologies, as described in Electric Power Research Institute (EPRI) guidance (EPRI, 2012), and RG 1.208 (NRC, 2007a). As an input, the staff used the Central Eastern United States Seismic Source Characterization (CEUS-SSC) model described in NUREG–2115 (NRC, 2012) along with the updated EPRI ground motion model (EPRI, 2013). Consistent with the EPRI guidance (EPRI, 2012), the NRC staff included all CEUS-SSC background seismic sources within a 500 km [310 mi] radius of the MURR site. In addition, the staff included all of the repeated large magnitude earthquake 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 a GMRS for comparison.

The control point is not specified in the safety analysis report (SAR) (University of Missouri, 2006). The biological shield of the reactor is supported by the concrete pad directly placed over the caisson. The caisson extends into the weathered rock layer. Therefore, the NRC staff assumed for this assessment that the control point is located at the top of the weathered limestone layer following the suggestions given in EPRI guidance (EPRI, 2012). The seismic motion would be estimated and compared at the control point. The GMRS also would be estimated at the same horizon.

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 MURR facility and the nearby nuclear power generating facility, the Callaway Nuclear Plant. Other available information relevant to site response analysis was also studied. Several boreholes were drilled to characterize the subsurface when constructing the containment building of the MURR facility (University of Missouri, 2006). Other boreholes were drilled later for surrounding facilities. These boreholes provide qualitative description of different soil strata beneath the containment building. However, these boreholes were too shallow to encounter the general rock conditions, defined as rock strata having shear wave velocity of 2.8 km/s [9,200 ft/s] (NRC, 2007a). Moreover, no shear wave velocity measurements are available in any of the subsurface materials at the MURR site. Only a few boreholes had the Standard Penetration Test (SPT) blow count measured at a few depth intervals but information on the blow hammer used in these tests is not available. Although the NRC staff estimated the shear wave velocity from the SPT results assuming a standard blow hammer, the data are incomplete for site response analysis because these boreholes were too shallow. However, descriptions of the soil strata at the MURR site match with those at the Callaway Nuclear Plant site (Ameren Missouri, 2014). There are only minor differences between the shear wave velocity for the soil layers at the Callaway Nuclear Plant site and those estimated at the MURR site from the SPT measurements. Consequently, the NRC staff has used in the site response analysis the shear wave velocity of each soil and rock stratum of the MURR site as available from the seismic assessment of the Callaway Nuclear Plant (Ameren Missouri, 2014). Figure 3.2-1 shows the shear wave velocity profile with depth from surface developed in this assessment.

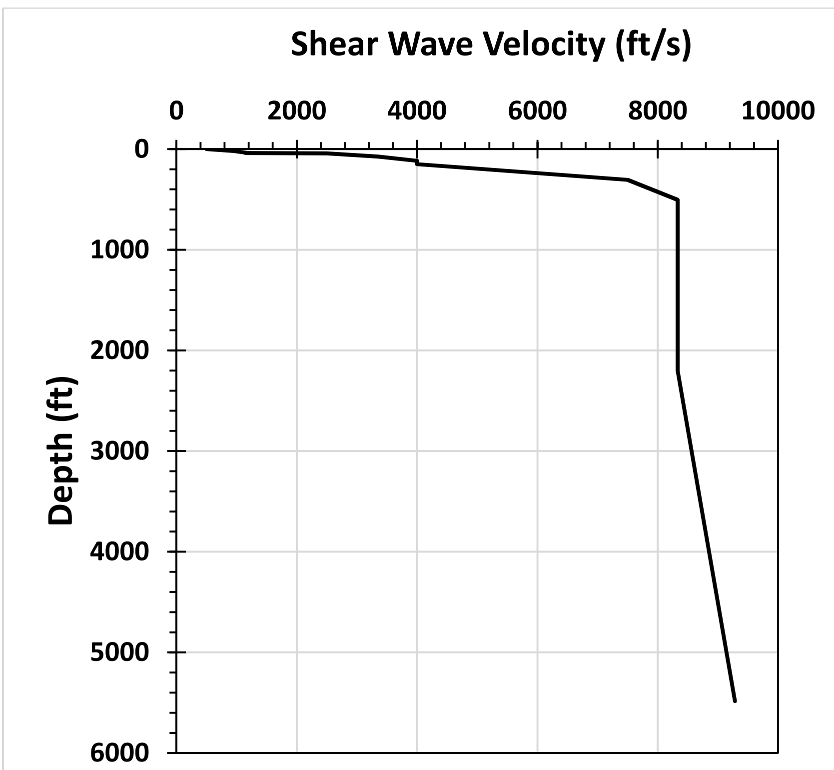

Figure 3.2-1. Shear Wave Profile for MURR site

No site-specific dynamic material properties are available either for the MURR facility or for the Callaway Nuclear Plant site. As used in the Callaway Nuclear Plant (Ameren Missouri 2014), the materials below the control point of the MURR site are represented by G/G_{max} and hysteretic damping values as given in EPRI rock curves and linear response for firm rock following EPRI guidance (EPRI, 2012). Peninsular Range G/G_{max} and hysteretic damping curves combined with linear response for firm rock are assumed to represent an equally plausible alternative response of the subsurface materials at the MURR site. Two different options of material properties have been used in this site response analysis. In option 1 (MURR7), both these sets of generic modulus reduction and hysteretic damping are considered to reflect a reasonable range in nonlinear dynamic material properties and associated nonlinearity in site response conditional on velocity profile, following EPRI guidance (EPRI, 2012). Both rock and soil curves have equal weight. As an alternate in this analysis, in option 2 (MURR11) only EPRI generic rock curves and hysteretic damping along with linear response of the firm rock are used.

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. For this assessment, kappa values used in Callaway Nuclear Plant assessment (Ameren Missouri, 2014) have been applied to this site. Callaway developed the base-case kappa using information from EPRI guidance (EPRI, 2012) for a firm rock Central and Eastern U.S. rock site. Three kappa values used in this analysis are 0.016, 0.020, and 0.009.

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 is developed from the base case profiles. Parameters developed by Toro (1997) for U.S. Geological Survey “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 top of the weathered rock layer. The results are shown in Figure 3.2-2. The NRC staff has also calculated the 10^{-4} and 10^{-5} uniform hazard response spectra using the results of its confirmatory PSHA and site response analyses and then computed the GMRS following the criteria in RG 1.208 (NRC, 2007a). The results are plotted in Figure 3.2-3. The GMRS is the performance-based site-specific ground motion response spectrum.

Figure 3.2-4 shows the seismic response spectrum with PGA of 0.2g, as used in the Callaway Nuclear Plant (Ameren Missouri, 2014). As discussed before, the MURR facility has adopted a similar seismic response spectrum as SSE. This figure also shows the GMRS estimated in this assessment for both options of material properties used (MURR7 and MURR11). First, there is practically no difference between these two options used for material properties on the GMRS estimated at the selected control point. Second, the GMRS is enveloped by the SSE up to about 16 Hz. The GMRS exceeds the SSE above this frequency. Based on EPRI guidance (EPRI, 2012), the ground motions at higher than approximately 10 Hz frequency are not damaging to the structures, systems, and components of a nuclear reactor except the functional performance of components sensitive to vibration, such as electrical relays. Based on the information provided during a phone call with the University of Missouri staff on June 14, 2016, every safety feature in the MURR facility has been designed to fail in “fail safe” mode. The relays would disengage due to high frequency vibration, similar to a loss of power event, resulting in dropped control rods in the reactor. Due to passive nature of the design, no electric power is needed for safe shutdown and maintaining cooling of the reactor. Therefore, based on this information, the NRC staff concludes that no additional assessment would be needed for the higher frequency ground motion exceedance at the MURR facility.

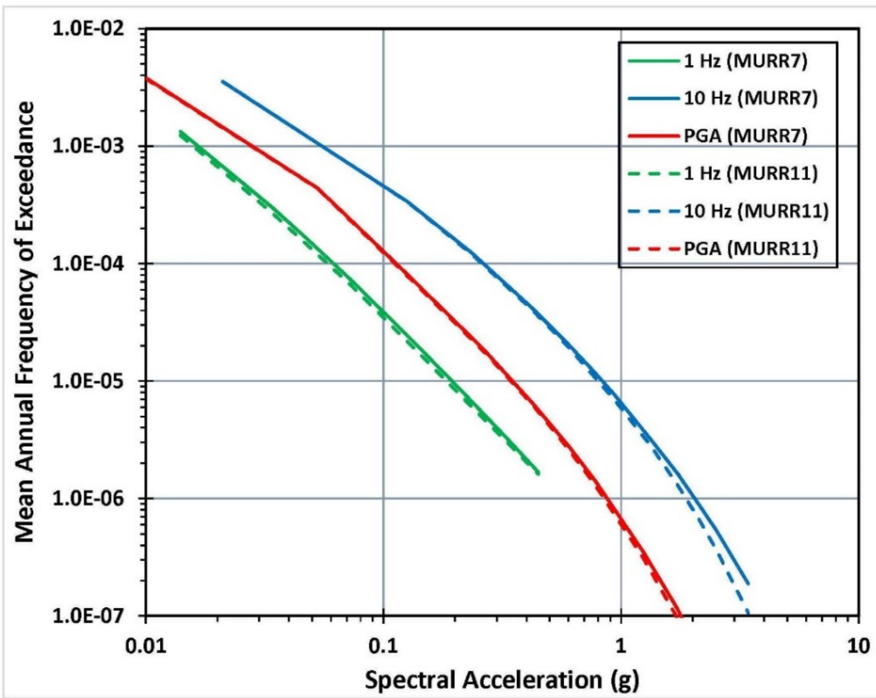

Figure 3.2-2 Mean Control Point Hazard Curves at Three Frequencies.

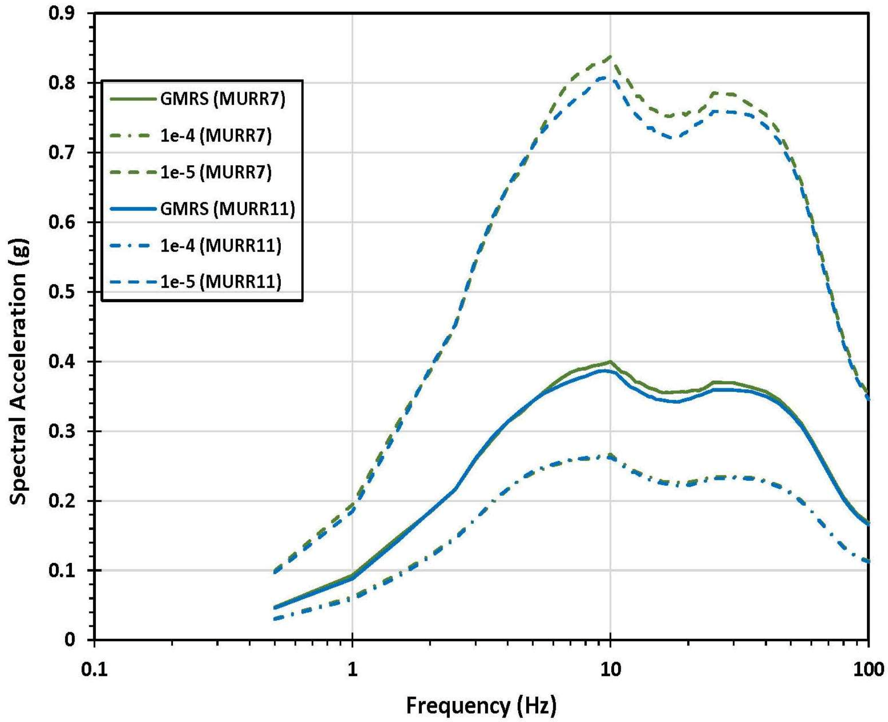

Figure 3.2-3. NRC Staff Estimated GMRS with 10^{-4} and 10^{-5} Uniform Hazard Spectra.

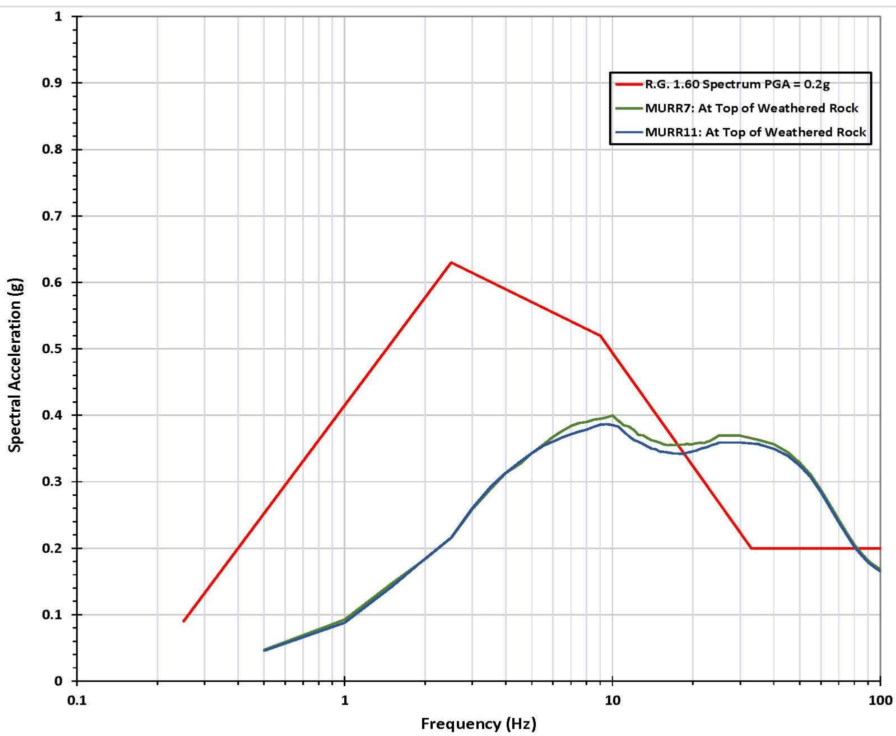

Figure 3.2-4. Comparison of the NRC Staff's GMRS with the Design Response Spectrum for the MURR Site

3.2.2.2 Effects of Sloshing from Seismic Event

The NRC staff has estimated the height of sloshing wave δ_s in the reactor pool due to a seismic event with a PGA of 0.2g using both ASCE 7–10 (ASCE, 2013) and EPRI guidance (EPRI, 2012) methodologies. The reactor pool is an aluminum-lined pool 3 m [10 ft] in diameter and 9.1 m [30 ft] in depth. The reactor vessel with the fuel is covered by 7.2 m [24 ft] of shielding water during normal operation (University of Missouri, 2006).

Because the pool is anchored to the biological shield at the side and at the bottom and open at the top, only overtopping of water due to sloshing induced by the seismic event is a potential concern. Damage from buckling of the wall or damage of wall from sloshing-induced forces is not a concern. Similarly, sliding of the pool or base uplift are also not potential concerns.

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 1.81 s. The spectral acceleration of the sloshing water with $T_c = 1.81$ s and 0.5% damping is estimated to be 0.35g using Equation (15.7–10).

The MURR 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 which pose a substantial hazard to the community and 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) and the estimated sloshing height, δ_s, would be 0.66 m [2.2 ft].

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 0.512 Hz or, alternatively, the period is 1.95 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 0.53 m [1.75 ft]. 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 0.63 m [2.1 ft], practically same as estimated using the ASCE 7-10 (ASCE, 2013) methodology.

The estimated sloshing height is smaller than the freeboard available in the pool (i.e., 1.9 m [6.3 ft]). Therefore, the NRC staff concluded that the available freeboard is sufficient to contain the potential sloshing effects from a seismic event defined by the response spectrum following RG 1.60 (NRC, 2014) with PGA 0.2g.

3.2.3 Assessment of Tornado-Missile Strikes

The NRC staff has assessed the potential for damage of the reactor core from a wind-related phenomenon at the MURR reactor site using current tornado information given in RG 1.76 (NRC, 2007b). The NRC staff has also referred to Kennedy (1976) for the modified formula of the National Defense Research Committee to assess the response of a concrete wall after a wind-generated missile strike. The missile is assumed rigid in this analysis for maximum penetration. It should be noted that at Columbia, Missouri, the location of the MURR 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 has selected a rigid large tornado missile, such as a Schedule 40 pipe, striking the exterior walls of the containment building for this assessment. This missile was selected as 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 41 m/s [135 ft/s]. Because the city of Columbia is located in Tornado Region I, a speed of 41 m/s [135 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 reactor core is at below grade (approximately 3.8 m [11.5 ft] below ground level) and surrounded by laboratory and office facilities on the first above-grade level (University of Missouri, 2006). A tornado missile has to perforate the 0.3-m [1-ft] thick reinforced concrete exterior wall and subsequently pass through the laboratories and support facilities surrounding the reactor inside the containment building before reaching the biological shield surrounding the reactor core. Striking the containment building above the first level is a much less likely scenario because the missile has to be lifted to a large height and propelled above the facilities before the strike.

Using the modified National Defense Research Committee formula for a missile striking a massive concrete wall (Kennedy, 1976), the Schedule 40 pipe striking in a direction perpendicular to the external concrete wall at a speed of 41 m/s [135 ft/s] would penetrate less than 0.13 m [5 in]. The estimated scabbing thickness measured from the interior side of the concrete barrier would be 0.5 m [1.7 ft]. A concrete thickness of approximately 0.6 m [2 ft] would be needed to prevent any scabbing at the interior wall. Additionally, perforation of the concrete wall would be prevented if the concrete has a thickness of at least 0.4 m [1.3 ft].

Based on the above assessment, a Schedule 40 pipe striking perpendicular to the exterior wall would be able to perforate it. However, in doing so, the missile will lose a substantial amount, if not all, of the kinetic energy so that it is unlikely to reach the biological shield at below grade and cause any significant damage to it after passing through the laboratories and support facilities. Even at the original speed of 41 m/s [135 ft/s], this missile would penetrate only 0.13 m [5 in] of the 0.9 to 2.3 m [3 to 7.5 ft]-thick biological shield. It should be noted that the missile speed of 41 m/s [135 ft/s] is associated with extremely rare tornadoes with strike frequency of 10^{-7} per year, appropriate for commercial power reactors (NRC, 2007b). The MURR reactor being a non-power research reactor with a thermal output of 10 MWt, the appropriate tornado for hazard assessment would be a more frequent one (i.e., a tornado with higher annual frequency of occurrence). The associated wind speed and impact speed of the wind-generated missiles will be substantially lower than 41 m/s [135 ft/s], assumed in this analysis. At a lower impact speed, the expected damage (penetration, scabbing, and perforation) of the exterior concrete wall by a tornado missile would be less severe. The missile may not even be able to perforate the exterior wall completely. Additionally, the reactor is located below grade, which provides additional protection from a tornado missile strike, not accounted for in this conservative hazard assessment.

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 that it is unlikely that the reactor at the MURR facility will experience any substantial damage from a rigid tornado missile strike even from an extremely rare one.

4.0 CONCLUSIONS

The NRC staff assessed the seismic hazard of the MURR facility using present-day methodologies (EPRI, 2012) and RG. Based on the re-evaluated seismic hazard, the GMRS is enveloped by SSE in the 1 to 16 Hz range. Although the GMRS exceeds the SSE at frequencies larger than 16 Hz, “fail safe” failure mode of the safety features will drop the control rods in the reactor in such events. Moreover, passive design of the reactor does not require electric power for safe shutdown. Therefore, the NRC staff concludes that no additional assessments would be needed for seismic-related hazards. Additionally, no other assessments would be needed for seismic-induced sloshing and high-wind-related hazards.

The NRC staff assessment concludes that current regulatory requirements for the MURR 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.

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