January 11, 2017

Dr. Thomas H. Newton, Deputy Director
National Institute of Standards and
Technology
NIST Center for Neutron Research
U.S. Department of Commerce
100 Bureau Drive, Mail Stop 6101
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SUBJECT: NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY– STAFF ASSESSMENT OF APPLICABILITY OF FUKUSHIMA LESSONS LEARNED TO THE NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY CENTER FOR NEUTRON RESEARCH TEST REACTOR

Dear Dr. Newton:

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 National Institute of Standards and Technology (NIST) Center for Neutron Research Test Reactor (NBSR). In a letter dated June 1, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15112A136), Dr. Robert Dimeo was informed of the NRC staff's intention to perform an audit of the NBSR 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 NBSR. The June 1, 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 NBSR 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 NBSR Docket No. 50-184. The NRC staff assessment concludes that current regulatory requirements for the NBSR 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. Xiaosong Yin, at (301) 415-1404 or by electronic mail at Xiaosong.Yin@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. 50-184
License No. TR-5

Enclosure:
As stated

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STAFF ASSESSMENT OF APPLICABILITY OF

FUKUSHIMA LESSONS LEARNED TO THE NATIONAL INSTITUTE OF STANDARDS

AND TECHNOLOGY CENTER FOR NEUTRON RESEARCH TEST REACTOR

LICENSE NO. TR-5; DOCKET NO. 50-184

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) staff identified the need for additional information for three high-power research and test reactors (RTRs) (including the National Institute of Standards and Technology Center for Neutron Research Test Reactor (NBSR)) 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 these assessments, NRC staff determined it was necessary to perform additional evaluations of the NBSR regarding the reactor’s capabilities to prevent or mitigate loss of coolant accidents (LOCAs) as a result of beyond-design-basis natural phenomena (e.g., seismic events, flooding, or high winds). For the NBSR, the early loss of reactor coolant can result in failure of the fuel cladding and subsequent radiological release unless reactor coolant makeup can be provided from installed facility equipment or from portable external sources.

The NBSR is licensed to operate at a maximum thermal power of 20 megawatts (MWt). In the case of the NBSR, if there is an extended loss of electrical power to operate the active decay heat removal systems or damage to the active decay heat removal system that prevents its use, fuel damage could occur unless the decay heat removal systems are restored or reactor coolant makeup using portable external sources can be deployed. As such, in addition to performing additional assessments regarding the licensee’s capabilities to prevent or mitigate LOCAs as a result of seismic, flooding or high winds events, the NRC staff stated in the March 2, 2015, preliminary assessment and in SECY 15-0081 that it would also assess the licensee’s capabilities to prevent or mitigate loss of decay heat removal capabilities as a result of an extended loss of all electrical power that could result from an extreme flood.

2.0 REGULATORY EVALUATION

The purpose of the NRC staff’s evaluation was to determine if additional regulatory action was necessary for the NBSR 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 as well as whether or not additional regulatory actions were necessary based on Fukushima lessons learned.

3.0 TECHNICAL EVALUATION

3.1 Applicability of Fukushima Lessons Learned to the NBSR

SECY 15-0081 provides a detailed evaluation of the applicability of Fukushima Lessons Learned to the NBSR. Specifically, SECY 15-0081, Enclosure 1, Section 8 states that the NBSR is protected initially from fuel cladding failure via a passive coolant makeup system combined with natural convection cooling following the loss of active decay heat removal capability. Specifically, the inner reserve tank will make up the inventory boiled off during the first half hour. The emergency tank will continue to replenish the inventory for an additional 2 hours. After this time the inventory of coolant contained in the passive makeup system becomes depleted. This could lead to the pool coolant level lowering due to boiling until the fuel becomes uncovered and fuel temperatures start to increase to a point where fuel cladding can fail. The lowering of the pool level due to boiling would be prevented if the facility's light water makeup water source to the core or electrical power and a means of decay heat removal are available or otherwise provided via portable equipment in a timely manner. This portable equipment can allow domestic light water to be supplied to the reactor following installation of a spool-piece. Emergency power to the active decay heat removal system can be supplied by the 125 volt direct current station batteries for a limited time or from one of the two on-site emergency diesel generators for a longer period of time. In the event of a LOCA, the heavy water can be collected in sumps and then pumped back into the reactor for decay heat removal;

however, this requires alternating current (AC) power. The test reactor fuel cladding is vulnerable to failure from beyond-design-basis natural phenomena events (e.g., seismic, flooding or high winds) that potentially result in an extended loss of electrical power, active decay heat removal systems, or coolant inventory makeup capability.

The radiological consequences resulting from a severe external event may exceed those assumed in 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," used in licensing the facility. This has been confirmed by the post-9/11 security assessment of sabotage scenarios which assumed massive damage states to the facility. Because of the malicious intent and the extreme assumptions of facility damage used in the sabotage assessment, the postulated radiological consequences from the worst case sabotage event are expected to bound the postulated radiological consequences of all external events. The radiological consequences predicted by the worst case sabotage event analysis are a fraction of the 10 CFR Part 100 reactor siting dose criteria[2].

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

3.2 <u>Staff Assessment of Potential for a Beyond Design Basis Natural Phenomena Event to Cause Core Damage at the NBSR</u>

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 Confinement Building and the Reactor

The NBSR facility is located within the NIST campus and is part of the NIST Center for Neutron Research. The NBSR is a heavy water moderated and cooled reactor with a tank-type design. The reactor is designed to operate at 20 MWt of power using highly enriched uranium fuel (NIST, 2004). Heavy water is the primary coolant that recirculates in a closed system.

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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".

[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.

The confinement building houses the reactor along with the spent fuel storage pool and the emergency cooling tank. The building is designed to confine the results of an accident. The exterior walls of the confinement building are 0.6 m [2 ft] thick and made of reinforced concrete. The reactor vessel is made of aluminum. It is 2 m [7 ft] in diameter and 5 m [16 ft] in height with an elliptical cap at the bottom and a flange at the top (NIST, 2004). A thermal shield is located in between the reactor vessel and the biological shield. The thermal shield protects the biological shield from excessive radiation heating. At the elevation of the reactor core, the thermal shield is made of 5 cm [2 in] thick lead and 20 cm [8 in] thick steel (NIST, 2004). The biological shield surrounds the thermal shield and rests on the concrete foundation. The biological shield is made of heavy magnetite concrete with a minimum dry density of 3,844 kg/m^3 [240 lb/ft^3]. The minimum thickness of the biological shield surrounding the reactor core at the reactor's high-flux central plane region is 1.9 m [6.2 ft]. The shield thickness reduces to 1.6 m [5 ft] at the lower floor level. The concrete was formed directly against the thermal shield of the core. The 28-day compressive strength of the concrete is 20.7 MPa [3,000 psi] (NIST, 2004).

3.2.2 Seismic Assessment Basis of the NIST Reactor Facility

A seismic event can result in a LOCA and concurrent loss of alternating current (AC) power to the NBSR facility. Assuming the 1897 Giles County, Virginia, earthquake with a Modified Mercalli Intensity of VIII (estimated as an m_b 5.8 earthquake) could occur at the NIST site (maximum potential earthquake), a maximum peak horizontal ground acceleration of 0.07 to 0.1g was estimated deterministically. No apparent distinction was made between earthquake ground motion in soil and in the bedrock (NIST, 2004; URS, 2003). This horizontal acceleration is in the range of a VII to VIII earthquake on the Modified Mercalli Intensity scale. The United States Geological Survey (USGS) seismic hazard maps of 2002 used a probabilistic approach incorporating smoothed seismicity and a large background zone. The USGS 2002 seismic hazard maps of the Central and Eastern United States (CEUS) show that estimated peak ground accelerations (PGAs) of 0.07g is appropriate with a 2,475-year return period (2 percent probability of exceedance in 50 years) for the NIST site (NIST, 2004; URS, 2003).

The confinement building was designed in accordance with the Building Officials and Code Administrators (BOCA) Codes for the area. The reactor vessel was also designed in accordance with the BOCA Codes for seismic loads (NIST, 2004). The confinement building and the reactor systems have been analyzed and shown to withstand the stresses generated by a 0.1g earthquake. The combined stress levels resulting from this earthquake in addition to all other design loads were well within the allowable limits for the various sections of the reactor vessel (NIST, 2004).

3.2.3 NRC Staff Confirmatory Seismic 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 safety analysis report (SAR) either presents only a PGA, as done for the NBSR facility, or refers to a nearby commercial reactor's Safe Shutdown Earthquake as their design ground motion.

The NRC staff has performed a probabilistic seismic hazard analysis (PSHA) for the NBSR facility site to assess the seismic safety of the NBSR facility using present-day methodologies.

As an input, the NRC staff has used the Central Eastern United States Seismic Source Characterization (CEUS-SSC) model in NUREG–2115 (NRC, 2012) along with the Updated Electric Power Research Institute (EPRI) ground motion model (EPRI, 2013). The NRC staff has included all CEUS-SSC background seismic sources within a 500 km [310 mi] radius of the NIST site. In addition, the NRC staff has 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 has used the resulting base rock seismic hazard curves together with a confirmatory site response analysis to estimate the PGA at the control point for comparison with the design ground motion.

The control point is not specified in the SAR (NIST, 2004). The NRC staff has assumed for this assessment that the control point is located at the bottom of the reactor foundation, which is taken at 12.6 m [42 ft] below the ground surface. The PGA of the ground motion would be estimated and compared at this control point location. Because the control point is located below the reactor foundation, all clayey and sandy materials of the subsurface, as observed in the boreholes, are not included in the site response analysis; only weathered rock and saprolite are included.

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 NBSR facility; nearby nuclear facilities, such as, North Anna and Calvert Cliff nuclear power stations; and other relevant studies conducted nearby. These studies includes damage to the Washington Monument from the Mineral, Virginia earthquake of 2011 (Kayen, et al., 2015; AMEC, 2012) and others (e.g., Waisnor, et al., 2001).

NIST is located in the Eastern Piedmont Physiographic province. This physiographic province is underlain by metamorphic rock formations with some igneous intrusions, generally consisting of gneisses and schists of the Precambrian age (URS, 2003; Waisnor, et al., 2001). The underlying rocks have been altered by physical and chemical weathering producing residual soil (saprolite). Saprolite in the Eastern Piedmont Physiographic province is difficult to characterize because of the unique development history. Weathering generally decreases with depth (Waisnor, et al., 2001). The subsurface profile shows a gradual transition from soil to decomposed-rock to rock with no well-defined boundaries (Waisnor, et al., 2001). These subsurface characteristics are also evident in the borehole logs from the NIST reactor site (NIST, 2004; URS, 2003). The soils generally consist of low plasticity micaceous clayey silts, sandy silts and silty sands with colors varying from yellow to brown to red brown.

URS (2003) provided the logs of several boreholes (Hole 1–A through 11–A) drilled by Burns and Roe at the NBSR facility during construction of the reactor in 1961. These logs include the description of different strata encountered in the boreholes in addition to the Standard

Penetration Test (SPT) blow counts. Coring of the rock was performed in these holes for several feet after the refusal of the SPT hammer. The shaley schist encountered in these holes was very seamy to seamy at this horizon with some amount of fragmentation. The fragmentation deceased substantially as the depth increased with increased proportion of core recovery.

The depth of bedrock is highly variable in the Piedmont province because of weathering of the bedrock (Waisnor, et al., 2001). USGS mapped the base of saprolite in Montgomery County, Maryland (Froelich, 1975). From this contour map, the approximate horizon of the saprolite-bedrock boundary has been estimated by the NRC staff at the NIST site. Based on the map, the contact between the base of the saprolite and the bedrock near the NBSR facility would be between 105 m and 120 m [350 and 400 ft] above the mean sea level (MSL), actually closer to the 105 m [350 ft] level contour. This estimated value fits well with the observation made in the boreholes (Hole 1–A through 11–A). Therefore, in this assessment, the bedrock is assumed to be at 107 m [355 ft] above MSL at the NBSR facility area.

No information is available on the shear wave velocity of the underlying subsurface materials at the NBSR reactor site (NIST, 2004; URS, 2003). The NRC staff has conducted an extensive literature search for information of the shear wave velocity measured in nearby areas. Available information is sparse and from locations far away from the NBSR facility. Additionally, the information is also from shallow depths not reaching far enough to encounter the general rock conditions, defined as rock layers having a shear wave velocity of 2.8 km/s [9,200 ft/s] (NRC, 2007a). As an alternative, the NRC staff has used empirical relationships available in the literature for converting the SPT blow counts to estimated shear wave velocities. For example, the relationship proposed by Seed, et al. (1983) have been used to estimate the shear wave velocities of sand, silty sand, and sandy silt layers from reported blow counts. Similarly, Rollins, et al. (1998) have been used to estimate the shear wave velocity of decomposed shaley schist assuming it is represented by gravel. Ohta and Goto (1978) have been used for fine sand layer with quartz and rock fragments assuming coarse sand would represent the material. Lee (1992) has been used to estimate the shear wave velocity of clayey silt. The estimated shear wave profile, calculated as average of the boreholes, is shown in Figure 3.2-1. This figure also show the shear wave velocity measured at the Reston, Virginia, Fire Station (Kayen, et al., 2015), approximately 24 km [15 mi] south of the NBSR site. Figure 3.2-1 shows that the estimated shear wave velocities at the NBSR facility are substantially higher than that measured at Reston Fire Station, although the general trend is similar. Lack of any other information from the NBSR site has precluded resolving the uncertainties associated with the estimated shear wave velocities at the NBSR facility.

As observed in the boreholes at the NBSR site, the strata below the saprolite-rock boundary are fragmented. The fragmentation decreases with depth. Therefore, the NRC staff has incorporated a transition zone with gradually increasing shear wave velocity at the NBSR facility because the fragmentation of the rock decreases with depth. However, the rate of shear wave increase and the depth at which the general rock condition would be encountered are unknown. The NRC staff has conducted a sensitivity study with two values of the parameters to assess how the estimated PGA is affected by varying the weathered zone thickness: (i) the general rock condition occurs at a depth of 60 m [200 ft] from the surface and (ii) the general rock condition occurs at 75 m [250 ft] from the surface.

The SAR (NIST, 2004) did not report any dynamic material properties. A search for the dynamic material properties from nearby sites did not produce any acceptable results. Measurements made at the Washington Monument site or Calvert Cliffs Nuclear Power Station are not used, as these sites are located in the coastal province with different subsurface materials. Shear modulus degradation curves and damping properties for same rock types (i.e., saprolite and weathered rock) are available from the North Anna nuclear power plant site (Virginia Electric and Power Company, 2015). Although the North Anna site is approximately 160 km [100 mi] south of the NBSR facility, the NRC staff has used these dynamic material properties for similar strata in the site response analysis.

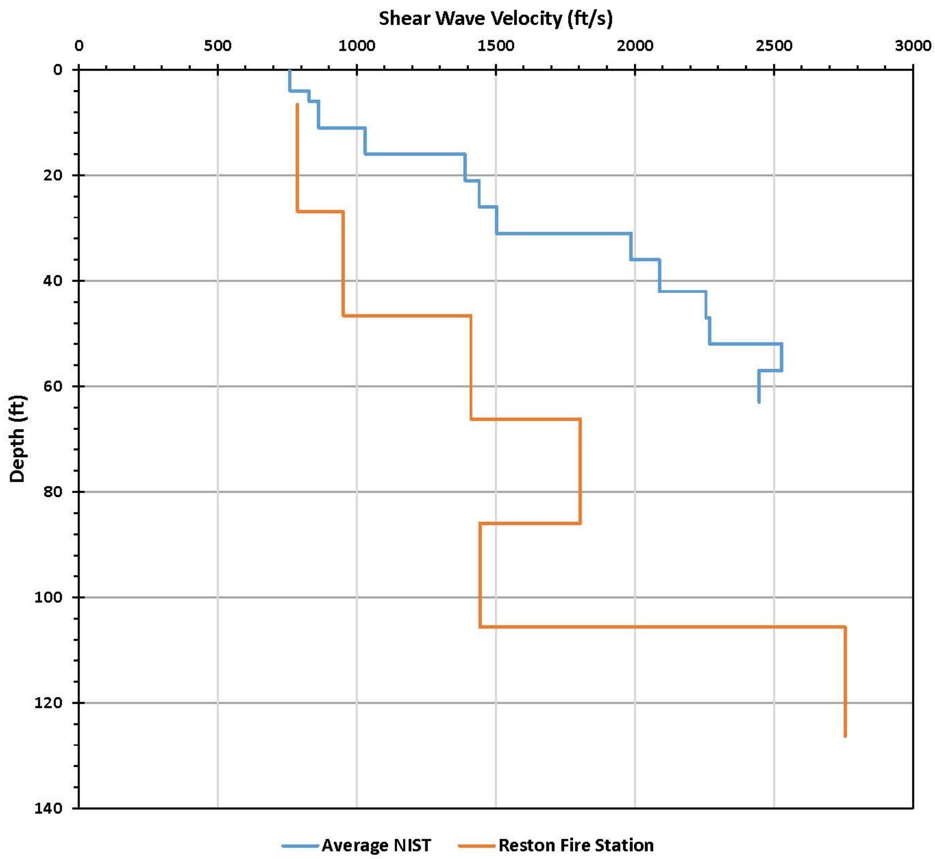

Figure 3.2-1. Estimated Shear Wave Velocity Profile with Depth at the NBSR Site. Measured Velocity at Reston, Virginia, Fire Station is Also Shown for Comparison.

The NRC staff has developed two sets of material properties for estimating the site response. In Set 1, the entire strata below the NBSR reactor is considered as composed of weathered rock, followed by the bedrock with general rock conditions. The weathered rock represents the transition zone with increasing shear wave velocity with depth and is modeled having the weathered rock properties, as estimated at the North Anna nuclear power plant site (Virginia Electric and Power Company, 2015). The bedrock with general rock conditions are represented as a linear material with no damping. In Set 2, the material below the reactor is assumed saprolite for up to 3.3 m [11 ft] followed by the weathered rock transitioning to the bedrock with general rock conditions. Material properties of saprolite, as measured at the North Anna nuclear power plant site, are used to model the saprolite at the NIST site. Other subsurface

materials are modeled as in Set 1. Both property sets are given equal weights in one simulation. The NRC staff has used several other combinations of the material properties to estimate their effects on the ground motion response spectrum (GMRS), for example, giving different weights to the material property sets. In addition to giving two sets equal weights, runs were done with either Set 1 or Set 2 with 80 percent weight. In one run, only weathered rock below the reactor has been assumed (i.e., only Set 1 materials have been used). Varying the material properties did not produce significantly different estimated GMRS. The estimated PGA was quite insensitive to variation of the material properties used in this assessment. Therefore, only one result is shown here for clarity.

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 guidance (EPRI, 2012). 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 seconds for the underlying hard rock. The estimated base-case kappa is 0.009 seconds. Epistemic uncertainty of kappa has been captured by two additional kappa values of 0.006 seconds and 0.015 seconds.

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 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 has 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 using different combinations of the material properties, as discussed before. The estimated mean hazard curves using material properties with weights of 20 percent for Set 1 and 80 percent for Set 2 are shown in Figure 3.2-2. Other combinations of the material properties produce results similar to these.

As discussed before, a PGA of 0.1g was used to design the NBSR facility (NIST, 2004). Based on the USGS seismic hazard map, the NIST has estimated the design earthquake would have a probability of exceedance of approximately 2 percent in 50 year or a return period of 2,475 year (NIST, 2004; URS, 2003). The NRC staff has assessed whether the reevaluated seismic hazard at the NBSR facility site creates a significantly higher seismic demand.

The NRC staff has calculated the PGA from the PSHA and site response analysis. The PGA is estimated to be approximately 0.07g from Figure 3.2-2 with a return period of 2,500 year (i.e., approximately 2 percent probability of exceedance in 50 year). The estimated PGA is 0.03g lower than the PGA reported in the SAR and used to design the NBSR facility (NIST, 2004). In addition, several design features of the reactor at the NBSR facility add additional safeguards against radiological consequences, as discussed in NRC (2015). Therefore, the NRC staff concludes that no additional assessment would be needed based upon this evaluation.

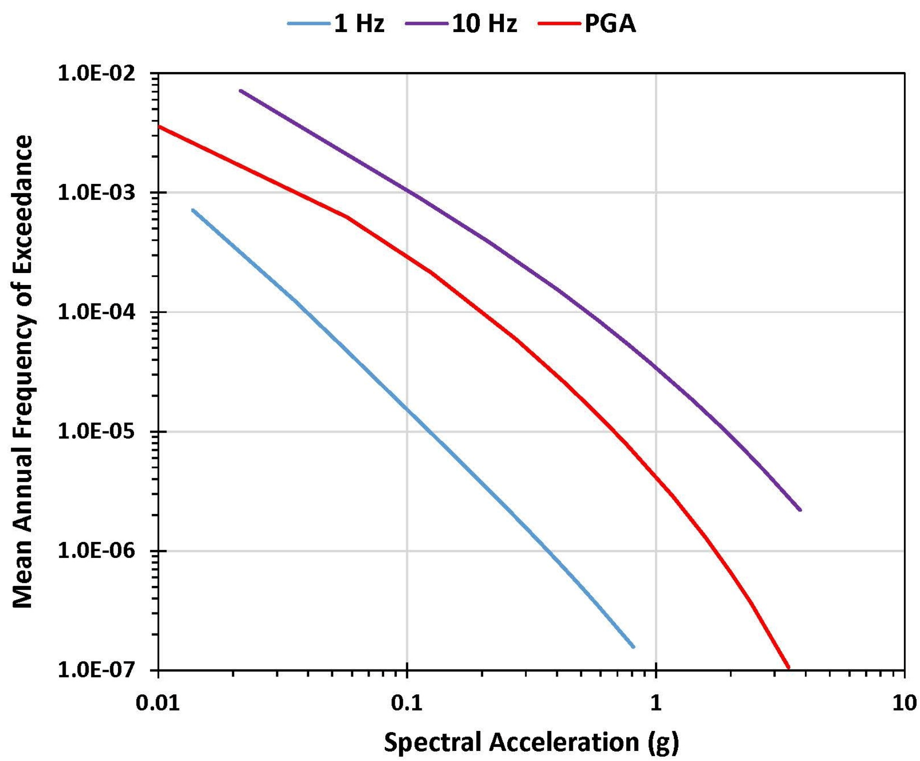

Figure 3.2-2. Mean Hazard Curves at the Control Point

The NRC staff has also assessed the potential for liquefaction of the subsurface materials during an earthquake following the guidance given in Regulatory Guide (RG) 1.198 (NRC, 2003). Two approaches have been used: (i) assess whether the physical characteristics, such as soil type and amount of fines, are conducive to liquefaction and (ii) assess the liquefaction susceptibility by estimating the factor of safety against liquefaction. For the second approach, the simplified procedure suggested by Youd, et al. (2001) has been used.

The amount of fines provided in the borehole logs from the NBSR reactor site is very high, much more than 30 percent (URS, 2003). Soils from the top portion of the subsurface have been classified as silt with low plasticity (e.g., soil group ML). Based on NRC (2003), the top portion of the subsurface is generally not liquefiable. Additionally, the thickness of the top soil and silt at the site are relatively thin. Based on that information, it is possible that no potential liquefaction hazard exists at the site.

In addition, the NRC staff has estimated the factor of safety against liquefaction using the SPT measurements from the boreholes. The SPT values have been standardized by applying different correction factors, following Youd, et al. (2001). Any effects of the fines have been corrected to compare with the seismic demand to estimate the factor of safety. The seismic demand as well as liquefaction resistance of each soil layer have been estimated. Additionally, the cyclic resistance ratio values have been compared with “Clean Sand Base” curve to assess

susceptibility to liquefaction. Results show that the potential for liquefaction at the NBSR site is negligible and confirms the conclusion made in SAR (NIST, 2004).

3.2.4 Effects of Sloshing from Seismic Events

The pool at the NBSR facility for storing spent fuel is rectangular with dimensions of 6.1 m [20 ft] length, 3 m [10 ft] width, and 5.5 m [18 ft] depth (NIST, 2004). A 15.2 cm [6 in] high curb is provided around the edge. The pool is kept normally covered. Therefore, the possibility of water spillage from sloshing effects exposing the spent fuel due to an earthquake is negligible.

3.2.5 Assessment of Tornado-Generated Missile Strikes

The NRC staff assessed the potential for damage to the reactor core from a wind-related phenomenon at the NBSR reactor facility using the RG 1.76 (NRC, 2007b). 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 reactor core after striking the confinement building. It should be noted that at Gaithersburg, Maryland, the location of the NBSR reactor facility, the expected speed of the tornado missiles are larger than the expected speed of any hurricane-generated missiles at the same annual frequency of exceedance (Vickery, et al., 2011). Therefore, the tornado missiles would be bounding in damage assessment from wind-generated missiles.

As discussed, the core of the test reactor at the NBSR facility is surrounded by a biological shield and is located inside the confinement building. The exterior walls of the confinement building and the biological shield are made of concrete (NIST, 2004). In addition, there are other rooms inside the confinement building that a tornado missile, after penetrating the outer wall, has to penetrate before reaching the reactor. The exterior walls of the confinement building are 0.6 m [2 ft] thick. The minimum thickness of the biological shield surrounding the reactor core in the reactor's high-flux central plane region is 1.9 m [6.2 ft]. The shield can be up to 1.6 m [5 ft] thick at the lower floor level. The concrete was formed directly against the thermal shield of the core. The 28-day compressive strength of the concrete is 20.7 MPa [3,000 psi] (NIST, 2004).

The NRC staff has estimated the effects of a rigid, large tornado missile, such as a Schedule 40 pipe, striking the confinement building. 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 a diameter 0.168 m [6.625 in] weighing 130 kg [227 lb], and traveling at a speed of 34 m/s [112 ft/s]. For this assessment, the impact speed of the missile was taken as 34 m/s [112 ft/s] because the NBSR facility is located in Tornado Region II (NRC, 2007b). It should be noted that this tornado missile speed is for a design-basis tornado having a wind speed with an exceedance frequency of 10^{-7} per year, appropriate for an operating nuclear power plant (NRC, 2007b). This is a conservative assumption because the reactor at the NBSR facility is not an operating power reactor. The heat output of the reactor is only 20 MWt.

Using the modified National Defense Research Committee formula (Kennedy, 1976), the NRC staff has estimated that a Schedule 40 pipe striking in a direction perpendicular to the

exterior wall of the confinement building at a speed of 34 m/s [112 ft/s] would penetrate approximately 0.10 m [0.3 ft]. The estimated scabbing thickness, measured from the interior side of the concrete wall, would be 0.50 m [1.7 ft]. A concrete wall having thickness of approximately 0.55 m [1.8 ft] would be needed to prevent any scabbing at the interior side of the wall. Additionally, perforation of the concrete wall would be prevented if the concrete has a thickness of at least 0.34 m [1.1 ft]. Because the exterior wall of the confinement building is 0.67 m [2.0 ft] thick and the reactor biological shield is located inside the confinement building, the hypothetical tornado missile would not be able to reach the biological shield even in a direct hit to the confinement building.

Effects of a crushable missile, such as an automobile, are expected to be much less severe, as most of the kinetic energy of the automobile would be absorbed by the confinement building exterior wall and only a minor fraction would be available for penetration. Therefore, based on the above discussion, the NRC staff concludes that the reactor at the NBSR facility would not experience any significant damage causing radioactive release to the public from a tornado missile strike at the exterior wall of the confinement building.

3.3 NRC Staff Assessment of Potential for a Beyond Design Basis Natural Phenomena Event Flood Event to Cause Loss of Decay Heat Removal at NBSR

The SAR described the NIST site as not being vulnerable to inundation by a probable maximum flood (PMF) event. This conclusion is based on the physical characteristics (primarily elevation) of the site relative to any surrounding water bodies. The SAR does not document a maximum water surface elevation for a PMF event.

The NRC staff visited the NIST site as part of an audit of the flood hazard review, as documented in the site audit plan issued on June 1, 2015 (ADAMS Accession No. ML15112A136). The site audit was conducted on July 16, 2015. During the audit, NRC staff examined the local site topography and storm drainage. In the front of the building, the first floor is above ground-level and is accessed by several steps. In the rear of the building, a roll-up door provides access to the walk-out basement. The roll-up door and basement entrance are flush with the ground, and NRC staff noted a potential for surface runoff to pond against the entrance.

Following Standard Review Plan Section 2.4, NRC staff considered the potential for site flooding due to a wide range of hazards. Based on the NRC staff's review of the watershed and the location of the NIST site far away from any ocean or large body of water, the only flooding mechanisms that could potentially inundate the NIST site are flooding from rivers and streams and flooding from local intense precipitation (LIP).

The site is located along a topographic ridge and at the headwater of a tributary to the Muddy Branch Creek. The nearest stream to Building 235 (reactor building) is an unnamed tributary of Muddy Branch Creek, approximately 1,000 ft [304.8 m] to the west-northwest of Building 235 (see Figure 1). Building 235 is more than 2000 ft [609.6] away from the main stem of the Muddy Branch Creek and more than 10 miles [16.1 km] upstream of the creek's confluence with the Potomac River. Thus, there is no credible potential for flooding of the site from backwater effects. The NRC staff also reviewed the most up-to-date Federal Emergency Management Agency flood insurance rate map and noted that the site is outside of the 500-year flood zone.

Therefore, based on this information, NRC staff concluded that inundation of the site due to river and stream flooding is not a credible flood hazard.

The only flood mechanism with the potential to impact the NIST site is flooding due to a LIP and associated site drainage. The NRC staff used a 1-hour, 1-square mile probable maximum precipitation (PMP) event to determine the potential for site flooding due to LIP (NUREG/CR-7046). The NRC staff also used a USGS topography map to determine the sub-basin drainage areas for the surface runoff flow-rate analysis. NRC staff also made the following conservative assumptions as part of the LIP review:

1. The rainfall depth is distributed uniformly over the NIST site (approximately 35 acres).
2. The estimated rainfall depth can be converted directly into surface runoff without loss using the Rational method (i.e., Runoff coefficient C-value of 1.0).
3. The flow depths, including overland flow and channel flow, can be described by the Manning equation.
4. The site's storm sewer system is not functioning.

Estimation of Local Intense Precipitation

National Oceanic and Atmospheric Administration Hydro-meteorological Reports (HMRs) were used by the NRC staff to calculate the 1-hour, 1-square mile, PMP depth. HMR No.51 provided the PMP for the 6-hour, 10-square mile storm at the site, which was then transformed into the 1-hour, 1-square mile, PMP by applying a conversion factor provided in HMR No. 52.

HMR No. 52, Figure 24, graphically shows the probable maximum precipitation for 1-hour, 1-square-mile storms, which for the NIST site was interpolated as 18.1 in [46 cm].

Surface Runoff

The NIST catchment area was divided into four sub-basins: DA1, DA2, DA3, and DA4 (see Figure 1). Sub-basins DA1 and DA2 contain the parking lot in front of Building 235 (reactor building). DA3 contains most of Building 235. The surface area of each sub-basin is shown on Table 1. The total catchment area is 32.54 acres [13.2 hectare] in Table 1, but in the flood hazard computations the total area is increased to 35 acres [14.2 hectare]. This increase was conservatively adopted to account for possible deviations due to estimating surface area.

Table 1 Sub-basin Areas

Sub-basin	Area (acres)
DA1	6.71
DA2	9.36
DA3	7.62
DA4	8.85
Total	32.54

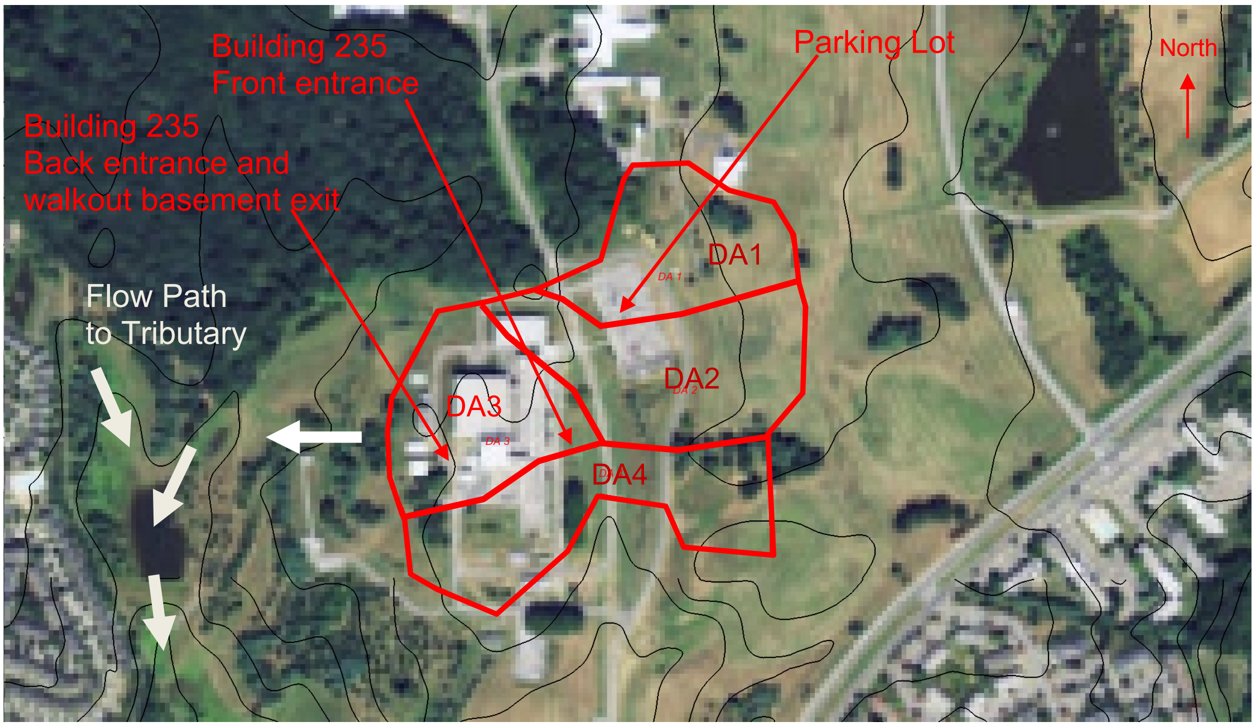

Figure 1 Drainage sub-basins, DA1, DA2, DA3, and DA4, are divided by red lines. Contours are shown in black lines. Upstream is on the right of the Building 235.

NRC staff calculated the maximum depth of water in the front entrance of Building 235 to be approximately 1 ft [30.5 cm]. Adjacent to the walk-out basement entrance door to Building 235, surface water runoff will pond. NRC staff's bounding calculations estimated the depth of ponding to be approximately 10 in [25.4 cm]. Ponded water would persist against the door for approximately 2 hours following cessation of the rainfall event.

Assessment of a LIP Event

If a LIP event were to disable the normal electric power source followed by a complete failure of all backup electric power sources to the facility, immediate cooling is provided by coast down of the primary pumps. After coast down, decay heat is then dissipated through natural convection that results in a gradual rise in reactor fuel and coolant temperature over several hours.

The licensee had performed a thermal hydraulic analysis using RELAP5 of this condition that shows that passive cooling as a result of natural convection is provided by the heavy water in the vessel, inner emergency cooling tank, and upper plenum is sufficient to provide adequate cooling to prevent failure of the reactor fuel cladding.

Alternative water sources to keep the core covered in the event of boiling losses include the volume of the inner heavy water emergency cooling tank and connections with a spool piece that could supply cooling water from either potable (main source) or fire main water (standby source), if available. During the site visit, the NRC staff learned that by using a special adaptor, the connection has the capability to connect to a fire truck as another alternate source of water.

Therefore, based on the above information, NRC staff concludes the NBSR facility would not experience any significant impact to NBSR's capabilities to prevent or mitigate loss of decay heat removal from a LIP event that impacts all electric power sources to the facility.

Flood Hazard Evaluation Conclusions

The NRC staff concludes that the PMF would not have a significant impact on the NIST reactor site. NRC staff examined all flooding mechanisms, and determined that the only hazard that could potentially challenge the site is ponding of runoff water following a LIP event.

The NRC staff assessed the ability of the NBSR to respond to an estimated LIP event using conservative assumptions. The NRC staff concludes that the LIP event would not have a significant impact to NBSR's capabilities to prevent or mitigate loss of decay heat removal resulting in a potential radioactive release to the public.

4.0 CONCLUSION

The NRC staff has assessed the seismic hazard of the NBSR facility using present-day information and methodologies. Based on the re-evaluated seismic hazard, the NRC staff concludes that no additional assessment would be needed for seismic hazards. Moreover, several design features of the reactor provide additional safeguards against radiological consequences resulting from a beyond design seismic event. Negligible potential for liquefaction would allow domestic light water to be supplied to the reactor using portable equipment for decay heat removal (NRC, 2015). In addition, the confinement building would be able to withstand tornado-generated missiles based on current information and guidance on a wind-related event.

Additionally, the NRC staff has assessed the flooding hazard of the NBSR facility using present-day information and methodologies. NRC staff concluded that inundation of the site due to river and stream flooding is not a credible flood hazard. Also, the NRC staff assessed a LIP event that could potentially impact the active cooling system and that passive heat removal is sufficient to protect the integrity of the reactor fuel cladding.

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

5.0 REFERENCES

Note: ADAMS Accession Nos. refer to documents available through NRC's Agencywide Document 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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