UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 26, 2015

Mr. Dennis L. Koehl
President and CEO/CNO
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - STAFF ASSESSMENT OF INFORMATION PROVIDED PURSUANT TO TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS* PART 50, SECTION 50.54(f), SEISMIC HAZARD REEVALUATIONS FOR RECOMMENDATION 2.1 OF THE NEAR-TERM TASK FORCE REVIEW OF INSIGHTS FROM THE FUKUSHIMA DAI-ICHI ACCIDENT (TAC NOS. MF3935 AND MF3936)

Dear Mr. Koehl:

On March 12, 2012, the U.S. Nuclear Regulatory Commission (NRC) issued a request for information pursuant to Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.54(f) (hereafter referred to as the 50.54(f) letter). The purpose of that request was to gather information concerning, in part, seismic hazards at each operating reactor site and to enable the NRC staff, using present-day NRC requirements and guidance, to determine whether licenses should be modified, suspended, or revoked.

By letter dated March 31, 2014, STP Nuclear Operating Company (the licensee), responded to this request for South Texas Project, Units 1 and 2 (STP). The NRC staff has reviewed the information provided related to the reevaluated seismic hazard for STP and, as documented in the enclosed staff assessment, determined that you provided sufficient information in response to Enclosure 1, Items (1) - (9) of the 50.54(f) letter.

The NRC staff concludes that the licensee responded appropriately and has completed its response to Enclosure 1, of the 50.54(f) letter. This closes out the NRC's efforts associated with TAC Nos. MF3935 and MF3936.

If you have any questions, please contact me at (301) 415-1617 or at Frankie.Vega@nrc.gov.

Sincerely,

Frankie Vega, Project Manager
Hazards Management Branch
Japan Lessons-Learned Division
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure:
Staff Assessment of Seismic
Hazard Evaluation and Screening Report

cc w/encl: Distribution via Listserv

STAFF ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO SEISMIC HAZARD AND SCREENING REPORT

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By letter dated March 12, 2012 (NRC, 2012a), the U.S. Nuclear Regulatory Commission (NRC or Commission) issued a request for information to all power reactor licensees and holders of construction permits in active or deferred status, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(f) "Conditions of license" (hereafter referred to as the "50.54(f) letter"). The request and other regulatory actions were issued in connection with implementing lessons-learned from the 2011 accident at the Fukushima Dai-ichi nuclear power plant, as documented in the "Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident" (NRC, 2011b).[1] In particular, the NRC Near-Term Task Force (NTTF) Recommendation 2.1, and subsequent Staff Requirements Memoranda (SRM) associated with Commission Papers SECY-11-0124 (NRC, 2011c) and SECY-11-0137 (NRC, 2011d), instructed the NRC staff to issue requests for information to licensees pursuant to 10 CFR 50.54(f).

Enclosure 1 to the 50.54(f) letter requests that addressees perform a reevaluation of the seismic hazards at their sites using present-day NRC requirements and guidance to develop a ground motion response spectrum (GMRS).

The required response section of Enclosure 1 requests that each addressee provide the following information:

(1) Site-specific hazard curves (common fractiles and mean) over a range of spectral frequencies and annual exceedance frequencies,

(2) Site-specific, performance-based GMRS developed from the new site-specific seismic hazard curves at the control point elevation,

(3) Safe Shutdown Earthquake (SSE) ground motion values including specification of the control point elevation,

(4) Comparison of the GMRS and SSE. A high-frequency evaluation (if necessary),

[1] Issued as an enclosure to Commission Paper SECY-11-0093 (NRC, 2011a).

Enclosure

(5) Additional information such as insights from NTTF Recommendation 2.3 walkdown and estimates of plant seismic capacity developed from previous risk assessments to inform NRC screening and prioritization,

(6) Interim evaluation and actions taken or planned to address the higher seismic hazard relative to the design basis, as appropriate, prior to completion of the risk evaluation (if necessary),

(7) Statement if a seismic risk evaluation is necessary,

(8) Seismic risk evaluation (if necessary), and

(9) Spent fuel pool (SFP) evaluation (if necessary).

Present-day NRC requirements and guidance with respect to characterizing seismic hazards use a probabilistic approach in order to develop a risk-informed performance-based GMRS for the site. Regulatory Guide (RG) 1.208, A Performance-based Approach to Define the Site-Specific Earthquake Ground Motion (NRC, 2007), describes this approach. As described in the 50.54(f) letter, if the reevaluated seismic hazard, as characterized by the GMRS, is not bounded by the current plant design-basis SSE, further seismic risk evaluation of the plant is merited.

By letter dated November 27, 2012 (Keithline, 2012), the Nuclear Energy Institute (NEI) submitted Electric Power Research Institute (EPRI) report "Seismic Evaluation Guidance: Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1 Seismic" (EPRI, 2012), hereafter called the SPID. The SPID supplements the 50.54(f) letter with guidance necessary to perform seismic reevaluations and report the results to NRC in a manner that will address the Requested Information Items in Enclosure 1 of the 50.54(f) letter. By letter dated February 15, 2013 (NRC, 2013b), the staff endorsed the SPID.

The required response section of Enclosure 1 to the 50.54(f) letter specifies that Central and Eastern United States (CEUS) licensees provide their Seismic Hazard and Screening Report (SHSR) by 1.5 years after issuance of the 50.54(f) letter. However, in order to complete its update of the EPRI seismic ground motion models (GMM) for the CEUS (EPRI, 2013), industry proposed a six-month extension to March 31, 2014, for submitting the SHSR. Industry also proposed that licensees perform an expedited assessment, referred to as the Augmented Approach, for addressing the requested interim evaluation (Item 6 above), which would use a simplified assessment to demonstrate that certain key pieces of plant equipment for core cooling and containment functions, given a loss of all alternating current power, would be able to withstand a seismic hazard up to two times the design-basis. Attachment 2 to the April 9, 2013, letter (Pietrangelo, 2013) provides a revised schedule for plants needing to perform (1) the Augmented Approach by implementing the Expedited Seismic Evaluation Process (ESEP) and (2) a seismic risk evaluation. By letter dated May 7, 2013 (NRC, 2013a), the NRC determined that the modified schedule was acceptable and by letter dated August 28, 2013 (NRC, 2013c), the NRC determined that the updated GMM (EPRI, 2013) is an acceptable GMM for use by CEUS plants in developing a plant-specific GMRS.

By letter dated April 9, 2013 (Pietrangelo, 2013), industry agreed to follow the SPID to develop the SHSR for existing nuclear power plants. By letter dated September 11, 2013 (Rencurrel, 2013), STP Nuclear Operating Company (the licensee) submitted at least partial site response information for South Texas Project, Units 1 and 2 (STP). By letter dated March 31, 2014 (Powell, 2014), the licensee submitted its SHSR.

2.0 REGULATORY BACKGROUND

The structures, systems, and components (SSCs) important to safety in operating nuclear power plants are designed either in accordance with, or meet the intent of Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 2: "Design Bases for Protection Against Natural Phenomena;" and Appendix A to 10 CFR Part 100, "Reactor Site Criteria." The GDC 2 states that SSCs important to safety at nuclear power plants shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

For initial licensing, each licensee was required to develop and maintain design bases that, as defined by 10 CFR 50.2, identify the specific functions that an SSC of a facility must perform, and the specific values or ranges of values chosen for controlling parameters as reference bounds for the design. The design bases for the SSCs reflect appropriate consideration of the most severe natural phenomena that had been historically reported for the site and surrounding area. The design bases also considered limited accuracy, quantity, and period of time in which the historical data have been accumulated.

The seismic design bases for currently operating nuclear power plants were either developed in accordance with, or meet the intent of GDC 2 and 10 CFR Part 100, Appendix A. Although the regulatory requirements in Appendix A to 10 CFR Part 100 are fundamentally deterministic, the NRC process for determining the seismic design-basis ground motions for new reactor applications after January 10, 1997, as described in 10 CFR 100.23, requires that uncertainties be addressed through an appropriate analysis such as a probabilistic seismic hazard analysis (PSHA).

Section 50.54(f) of 10 CFR states that a licensee shall at any time before expiration of its license, upon request of the Commission, submit written statements, signed under oath or affirmation, to enable the Commission to determine whether or not the license should be modified, suspended, or revoked. On March 12, 2012, the NRC staff issued requests for licensees to reevaluate the seismic hazards at their sites using present-day NRC requirements and guidance, and identify actions planned to address plant-specific vulnerabilities associated with the updated seismic hazards.

Attachment 1 to Enclosure 1 of the 50.54(f) letter described an acceptable approach for performing the seismic hazard reevaluation for plants located in the CEUS. Licensees are expected to use the CEUS Seismic Source Characterization (CEUS-SSC) model in NUREG-2115 (NRC, 2012b) along with the appropriate EPRI (2004, 2006) ground motion models. The SPID provided further guidance regarding the appropriate use of GMMs for the CEUS. Specifically, Section 2.3 of the SPID recommended the use of the updated GMM (EPRI, 2013) and, as such, licensees used the NRC-endorsed updated EPRI GMM instead of the older EPRI

(2004, 2006) GMM to develop PSHA base rock hazard curves. Finally, Attachment 1 requested that licensees conduct an evaluation of the local site response in order to develop site-specific hazard curves and GMRS for comparison with the plant SSE.

2.1 Screening Evaluation Results

By letter dated March 31, 2014 (Powell, 2014), the licensee provided its SHSR for the STP site. The licensee's SHSR indicated that the plant GMRS was bounded by the SSE for STP over the frequency range of 1 to 10 Hertz (Hz). As such, STP screens out to perform a seismic risk evaluation and SFP evaluation. The GMRS is also bounded by the SSE at frequencies above 10 Hz. The licensee indicated that a high-frequency confirmation will not be performed.

On May 9, 2014 (NRC, 2014), the NRC staff issued a letter providing the outcome of its 30-day screening and prioritization evaluation. As indicated in the letter, the NRC staff confirmed the licensee's screening results. The licensee's GMRS, as well as the staff's confirmatory GMRS, is bounded by the SSE over the frequency range of 1 to 100 Hz. Therefore, a seismic risk evaluation, SFP evaluation, and a high-frequency confirmation are not merited for STP.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's submittal to determine if the provided information responded appropriately to Enclosure 1 of the 50.54(f) letter with respect to characterizing the reevaluated seismic hazard.

3.1 Plant Seismic Design-Basis

Enclosure 1 of the 50.54(f) letter requests the licensee provide the SSE ground motion values, as well as the specification of the control point elevation(s) for comparison to the GMRS. For operating reactors licensed before 1997, the SSE is the plant licensing basis ground motion and is characterized by 1) a peak ground acceleration (PGA) value which anchors the response spectra at high frequencies (typically at 33 Hz for the existing fleet of Nuclear Power Plants); 2) a response spectrum shape which depicts the amplified response at all frequencies below the PGA; and 3) a control point where the SSE is defined.

In Section 3.1 of its SHSR, the licensee described its seismic design-basis for STP. The licensee stated that the design-basis of STP was developed as specified in 10 CFR Part 100, Appendix A, based on the maximum earthquake potential in the site region. The licensee identified a maximum earthquake of intensity VI at the site produced by either an intensity VII earthquake occurring 70 miles (110 km) away or an intensity VI earthquake adjacent to the site at a depth of 34,000 ft. (10,400 m). The response spectrum for STP is anchored at 0.1 g (10 percent the acceleration of earth's gravity) with a Regulatory Guide 1.60 spectral shape. The licensee specified that the control point is located at the ground surface.

Based on its review of the licensee's submittal and the updated final safety analysis report (UFSAR)(UFSAR, STP, 2010), the NRC staff confirms that the licensee's SSE, as well as the SSE control point elevation, are consistent with information provided in the STP UFSAR, as well as guidance in the SPID.

3.2 Probabilistic Seismic Hazard Analysis

In Section 2.2 of its SHSR, the licensee stated that, in accordance with the 50.54(f) letter and the SPID, it performed a PSHA using the CEUS-SSC model and the updated EPRI GMM for the CEUS (EPRI, 2013). The licensee used a minimum magnitude cutoff of **M**5.0, as specified in the 50.54(f) letter. The licensee further stated that it included the CEUS-SSC background sources out to a distance of 400 mi (640 km) around the site and included the Commerce, Eastern Rift Margin-North, Eastern Rift Margin-South, Marianna, Meers, and New Madrid Fault System Repeated Large Magnitude Earthquake (RLME) sources, which lie within 620 mi (1,000 km) of STP. The RLME sources are those source areas or faults for which more than one large magnitude (**M** ≥ 6.5) earthquake has occurred in the historical or paleo-earthquake (geologic evidence for prehistoric seismicity) record. The licensee used the Gulf version of the updated EPRI GMM for each of the CEUS-SSC sources. Consistent with the SPID, the licensee provided its base rock seismic hazard curves since the licensee used Method 2a to develop input ground motions for a site response analysis to determine the control point seismic hazard curves. The licensee provided its control point seismic hazard curves in Section 2.3.7 of its SHSR. The NRC staff's review of the licensee's control point seismic hazard curves is provided in Section 3.3 of this staff assessment.

As part of its confirmatory analysis of the licensee's GMRS, the NRC staff performed PSHA calculations for base rock site conditions at the STP site. As input, the NRC staff used the CEUS-SSC model as documented in NUREG-2115 (NRC, 2012b) along with the updated EPRI GMM (EPRI, 2013). Consistent with the guidance provided in the SPID, the NRC staff included all CEUS-SSC background seismic sources within a 310 mi (500 km) radius of the STP site. In addition, the NRC staff included the Eastern Rift Margin-South, Marianna, Meers, and New Madrid Fault System RLME sources, which lie within 620 km (1,000 mi) of the STP site. For CEUS-SSC sources with source to site paths that predominantly crossed the midcontinent region, the staff used the Midcontinent version of the updated EPRI GMM and for sources with paths that predominantly crossed the gulf region, the NRC staff used the Gulf version of the updated EPRI GMM.

Figure 3.2-1 of this assessment shows a comparison of the licensee's and the staff's base rock seismic hazard curves at 1, 10, and 100 (PGA) Hz. As shown in Figure 3.2-1, the licensee's and the NRC staff's base rock hazard curves are similar, which indicates that differences in licensee's and staff's PSHA inputs had little impact on the results.

Based on its review of the SHSR, the NRC staff concludes that the licensee appropriately followed the guidance provided in the SPID for selecting the PSHA input models and parameters for the site. This includes the licensee's use and implementation of the CEUS-SSC model and the updated EPRI GMM.

3.3 Site Response Evaluation

After completing PSHA calculations for reference rock conditions, Attachment 1 to Enclosure 1 of the 50.54(f) letter requests that the licensee provide a GMRS developed from the site-specific seismic hazard curves at the control point elevation. In addition, the 50.54(f) letter specifies that the subsurface site response model, for both soil and rock sites, should extend to sufficient depth to reach the generic or reference rock conditions, as defined in the GMMss used in the PSHA. To develop site-specific hazard curves at the control point elevation, Attachment 1 requests that the licensee perform a site response analysis.

Detailed site response analyses were not typically performed for many of the older operating plants; therefore, Appendix B of the SPID provides detailed guidance on the development of site-specific amplification factors (including the treatment of uncertainty) for sites that do not have detailed, measured soil and rock parameters to extensive depths.

The purpose of the site response analysis is to determine the site amplification that would occur as a result of bedrock ground motions 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 these layers, and the degree to which the shear modulus and damping change with increasing input bedrock amplitude. To develop site-specific hazard curves at the control point, the licensee performed a site response analysis.

3.3.1 Site Base Case Profiles

In its SHSR, the licensee indicated that it performed a site response analysis for STP. According to the licensee, the current site grade of Elevation 28 ft. (8.5 m) is underlain by approximately 34,500 ft. (10,500 m) of Coastal Plain sediments before Mesozoic basement rock is encountered. The uppermost materials are 750 ft. (228 m) of sediments of the Beaumont formation, beneath which are soil and soft rock deposits of increasing age to a depth of about 4,400 ft. (1,340 m) at which point the Oakville Sandstone is encountered to a depth of 6,200 ft. (1,900 m). Between the Oakville Sandstone and the Mesozoic basement rock is Cretaceous bedrock.

The licensee provided site profile descriptions in Sections 2.3.1 and 2.3.2 of its SHSR based on information in the STP UFSAR (STP, 2010). The licensee noted that the reactor containment buildings are founded on dense to very dense sand at 60 ft. below plant grade. The licensee used the velocity measurements from the UFSAR, as well as those from recent Combined License (COL) investigations for proposed Units 3 and 4, which consist of cross-hole, suspension P-S logging, and oil well log data. The licensee developed two different base cases for the STP site, one using the average Vs from the Units 1 and 2 investigations to a depth of 341 ft. (104 m) and the second using the stratigraphy of Units 1 and 2 but the measured Vs from Units 3 and 4 to a depth of 341 ft. (104 m). Below 341 ft. (104 m), both profiles are the same. The licensee developed upper and lower base case profiles using a logarithmic standard deviation of 0.20 in the upper 530 ft. (162 m), 0.19 between 530 and 603 ft. (162 and 184 m) and, below that depth, the upper and lower bound shear-wave velocities are based on the standard deviation of all data collected in each 200 ft. (31 m) interval).

To model the potential nonlinear behavior of the soils due to dynamic loading, the licensee used the laboratory testing for Units 3 and 4 to select appropriate shear modulus and damping curves. Based on a comparison of the data from the laboratory testing for Units 3 and 4, the licensee selected the EPRI (1993) curves and the Vucetic and Dobry curves (1991) for the sand, clay and silt layers beneath the site. Below a depth of 603 ft. (184 m), the licensee assumed linear behavior for the soils and used kappa estimates to account for the strain-independent damping ratios.

The licensee also considered the impact of kappa, or small strain damping, on site response. Kappa is measured in units of seconds (sec), and is the damping contributed by both intrinsic hysteretic damping, as well as scattering due to wave propagation in heterogeneous material. The licensee stated that for a deep soil site, like STP, a median kappa of 0.04 sec is appropriate for both base case profiles. To estimate kappa values for the lower and upper base case profiles, the licensee used a natural log standard deviation of 0.4, resulting in kappa values of 0.024 sec and 0.067 sec, respectively, for both base case profiles. The licensee weighed each possible base case profile, as shown in Table 2.3.2-8 of the SHSR.

To account for aleatory variability in material properties across the plant site in its site response calculations, the licensee stated that it randomized its base case profiles in accordance with Appendix B of the SPID. For the profiles with reference rock at a depth of 34,500 ft. (10,500 m), the licensee stated that it truncated the soil column at a best estimate depth of 8,094 (2,467 m) which reflects a soil column frequency of less than 0.1 Hz.

3.3.2 Site Response Method and Results

In Section 2.3.4 of its SHSR, the licensee stated that it followed the guidance in Section 2.5.3 of the SPID to develop input ground motions for the site response analysis, and in Section 2.3.5 of its SHSR, the licensee described its implementation of the random vibration theory (RVT) approach to perform its site response calculations. Finally, Section 2.3.6 of the SHSR shows the resulting amplification functions and associate uncertainties for the eleven input loading levels for the each base case profile.

In order to develop probabilistic site-specific control point hazard curves, as requested in Requested Information Item (1) of the 50.54(f) letter, the licensee used Method 2A, described in Appendix B-6.0 of the SPID. The licensee's use of Method 2A involved computing the site-specific control point elevation hazard curves for a broad range of spectral accelerations by combining the site-specific bedrock hazard curves, determined from the initial PSHA (Section 3.2 of this assessment), and their associated uncertainties, and the site-specific estimates of soil or soft-rock response determined from the site response analysis.

3.3.3 Staff Confirmatory Analysis

To confirm the licensee's site response analysis, the NRC staff performed site response calculations for the STP site. The NRC staff independently developed a shear-wave velocity profile, damping values, and modeled the potential nonlinear behavior of the site using measurements and geologic information provided in the STP, Units 1 and 2 UFSAR (STP, 2010), the STP, Units 3 and 4 FSAR (STP, 2014), and Appendix B of the SPID. For its site response calculations, the NRC staff employed the RVT approach and developed input ground motions in accordance with Appendix B of the SPID.

Because the site is well characterized and the velocities reported in the UFSAR for STP, Units 1 and 2 and proposed STP, Units 3 and 4 are similar, the NRC staff used only a single shear wave velocity profile rather than the two profiles used by the licensee. For the depth interval between the control point and the base of the profile, the NRC staff used the shear-wave velocity profiles in the FSAR for the proposed STP, Units 3 and 4 (STP, 2014). This profile is based on the recent data collected at the STP site and is consistent with data from the STP Units 1 and 2 UFSAR (STP, 2010), indicating that the site is relatively uniform. Figure 3.3-1 of this assessment shows the NRC staff velocity profile compared to the base case profiles developed by the licensee.

Similar to the approach used by the licensee, the NRC staff used information from the STP, Units 3 and 4 FSAR (STP, 2014) to select the damping and shear modulus reduction profiles for the site. The NRC staff used the Vucetic and Dobry (1991) curves for clay layers and the EPRI (1993) curves for sand layers for the upper 603 ft. (184 m). Below this depth, the NRC staff assumed linear behavior and used kappa to account for strain-independent damping. Following guidance in the SPID, the NRC staff assumed a site kappa of 0.04 sec for the STP site. The NRC staff also considered epistemic uncertainty in kappa using a natural log standard deviation of 0.4, resulting in kappa values that ranged from 0.028 sec to 0.05 sec.

Figure 3.3-2 of this assessment shows a comparison of the licensee's and NRC staff's median site amplification functions and uncertainties (±1 standard deviation) for two input loading levels. Amplification functions developed by the NRC staff are generally higher than those developed by the licensee across the frequency range of interest. However, these differences are small and are do not substantially effect the hazard results as shown below in Figure 3.3-3, which shows the licensee's and staff's control point hazard curves for STP.

Appendix B of the SPID provides guidance for performing site response analyses, including capturing the uncertainty for sites with less subsurface data; however, the guidance is neither entirely prescriptive nor comprehensive. As such, various approaches in performing site response analyses, including the modeling of uncertainty, are acceptable.

In summary, the NRC staff concludes that the licensee's site response was conducted using present-day guidance and methodology, including the NRC-endorsed SPID. The NRC staff performed independent calculations which confirmed that the licensee's amplification factors and control point hazard curves adequately characterize the site response, including the uncertainty associated with the subsurface material properties, for the STP site.

3.4 Ground Motion Response Spectra

In Section 2.4 of its SHSR, the licensee stated that it used the control point hazard curves, described in SHSR Section 2.3.7, to develop the 10^{-4} and 10^{-5} (mean annual frequency of exceedance) uniform hazard response spectra (UHRS) an then computed the GMRS using the criteria in RG 1.208.

The NRC staff independently calculated the 10^{-4} and 10^{-5} UHRS using the results of its confirmatory PSHA and site response analysis, as described in Sections 3.2 and 3.3 of this staff assessment, respectively. Figure 3.4-1 of this assessment shows a comparison of the GMRS determined by the licensee to that determined by the NRC staff. As shown in Figure 3.4-1, the licensee's GMRS shape is very similar to that calculated by the NRC staff.

The NRC staff confirms that the licensee used the present-day guidance and methodology outlined in RG 1.208 and the SPID to calculate the horizontal GMRS, as requested in the 50.54(f) letter. The NRC staff performed both a PSHA and site response confirmatory analysis and achieved results consistent with the licensee's horizontal GMRS. As such, the staff concludes that the GMRS determined by the licensee adequately characterizes the reevaluated hazard for the STP site.

4.0 CONCLUSION

The NRC staff reviewed the information provided by the licensee for the reevaluated seismic hazard for the STP site. Based on its review, the NRC staff concludes that the licensee conducted the seismic hazard reevaluation using present-day methodologies and regulatory guidance, appropriately characterized the site given the information available, and met the intent of the guidance for determining the reevaluated seismic hazard. Based on the preceding analysis, the NRC staff concludes that the licensee provided an acceptable response to Requested Information Items (1) – (9), identified in Enclosure 1 of the 50.54(f) letter.

In reaching this determination, the NRC staff confirms the licensee's conclusion that the licensee's GMRS for the STP site is bounded by the SSE in the 1 to 100 Hz frequency range. As such, a seismic risk evaluation, SFP evaluation, and high frequency confirmation are not merited. Based upon the preceding analysis, the NRC staff concludes that the licensee responded appropriately and has completed its response to Enclosure 1, of the 50.54(f) letter.

REFERENCES

Note: ADAMS Accession Nos. refers 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.

U.S. Nuclear Regulatory Commission Documents and Publications

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Figure 3.2-1 Plot of Staff's and Licensee's Base Rock Mean Hazard Curves at a Variety of Frequencies for the South Texas Project site

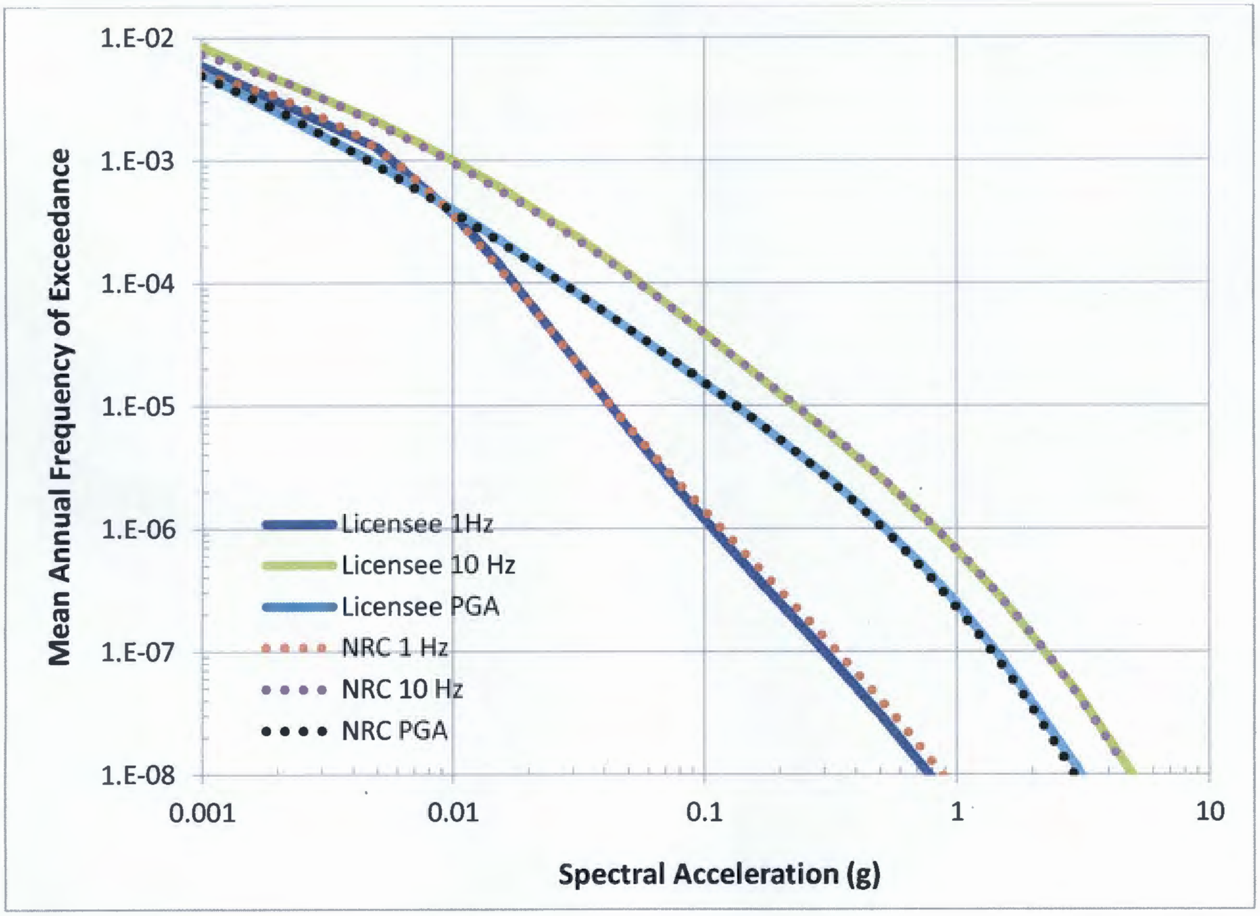


Figure 3.3-1 Plot of Staff's and Licensee's Base Case Shear-Wave Velocity Profiles for the South Texas Project site

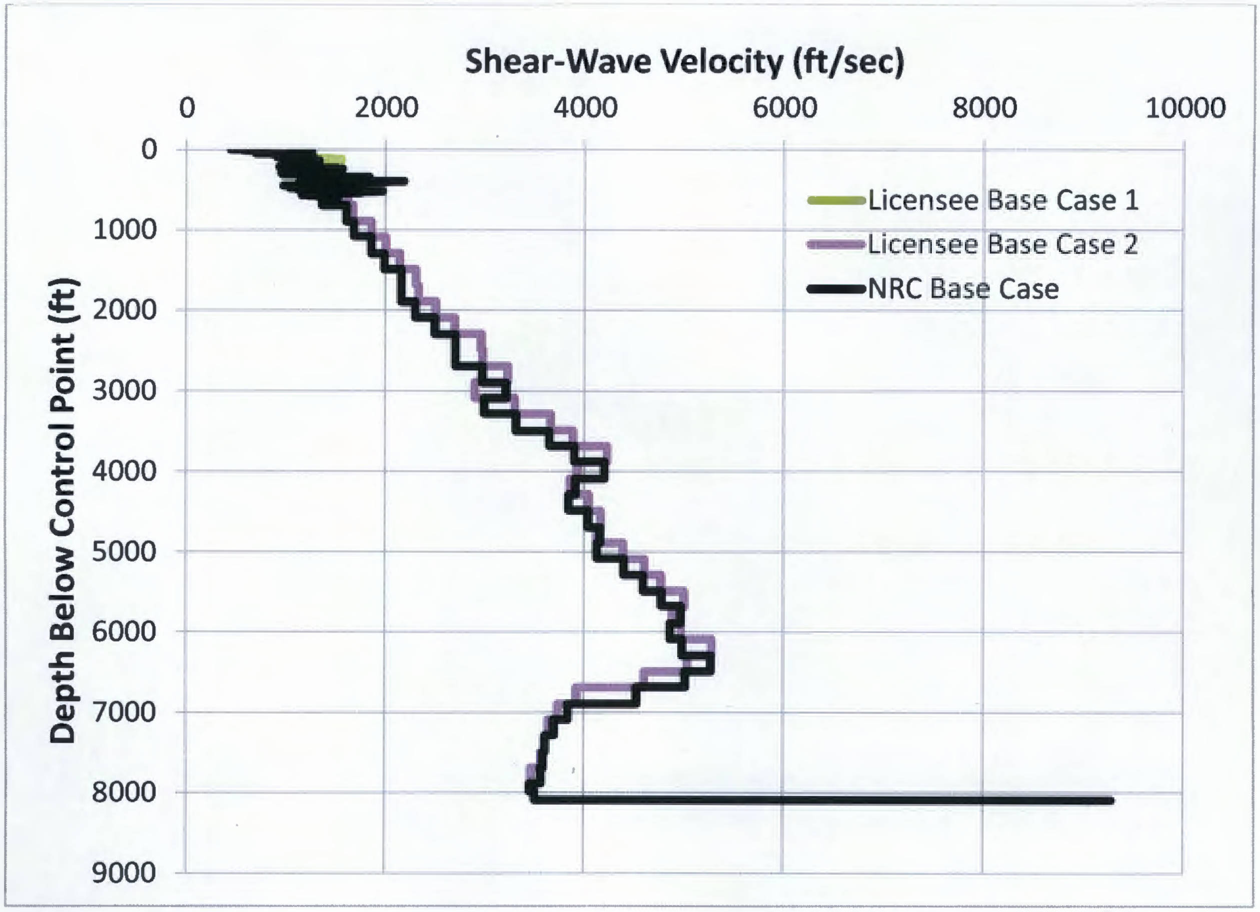


Figure 3.3-1 Plot Comparing the Staff's and the Licensee's Median Amplification Functions and Uncertainties for the South Texas Project site

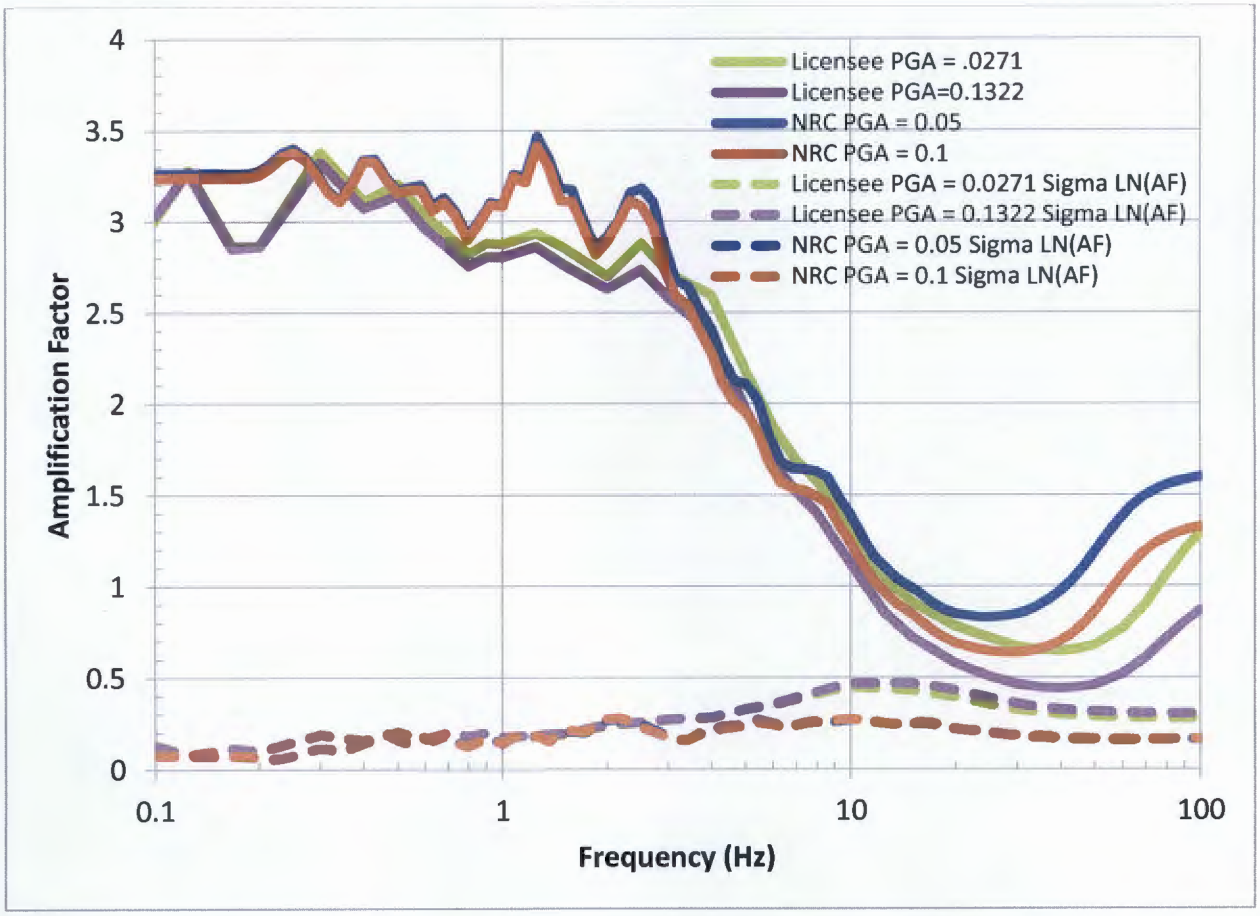


Figure 3.3-2 Plot Comparing the Staff's and the Licensee's Mean Control Point Hazard Curves at a Variety of Frequencies for the South Texas Project site

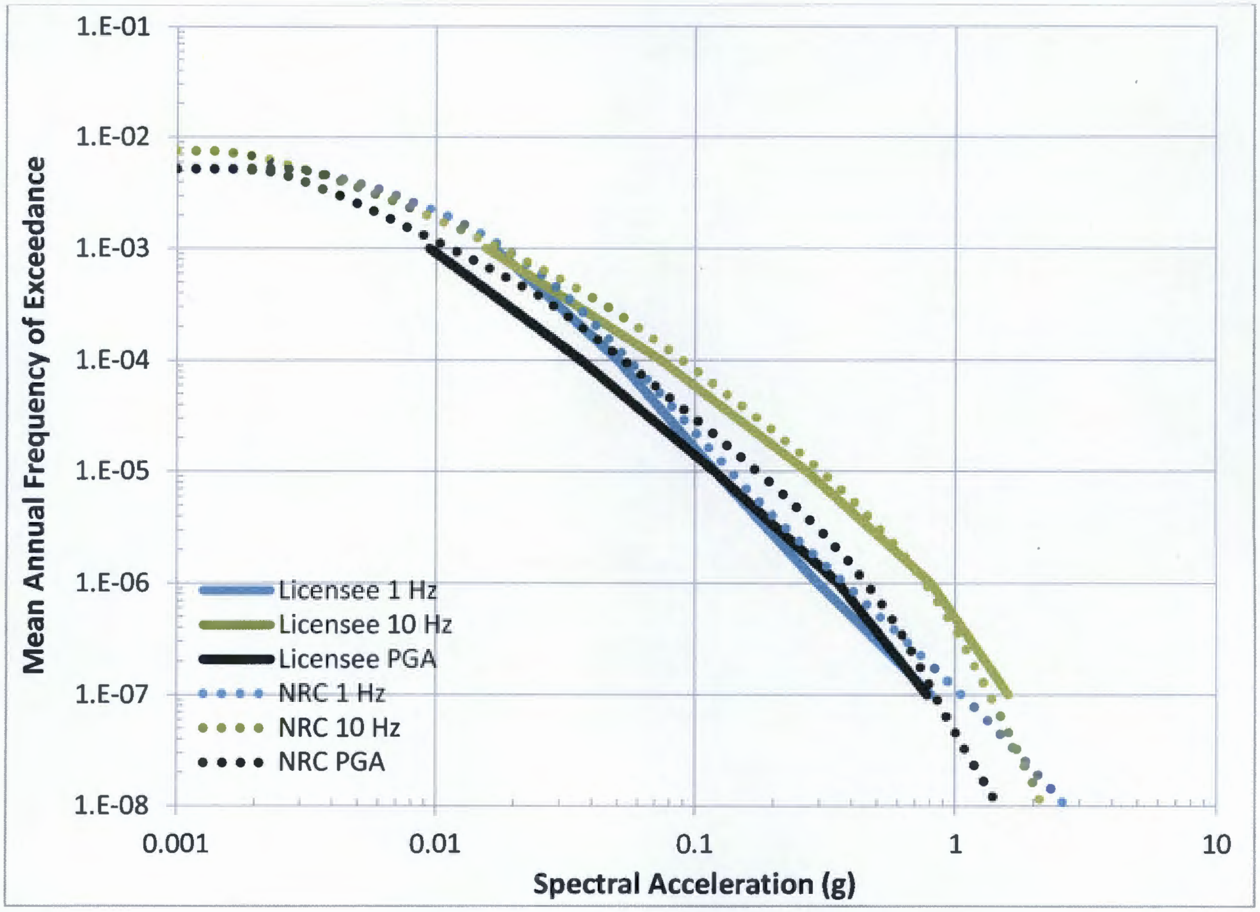
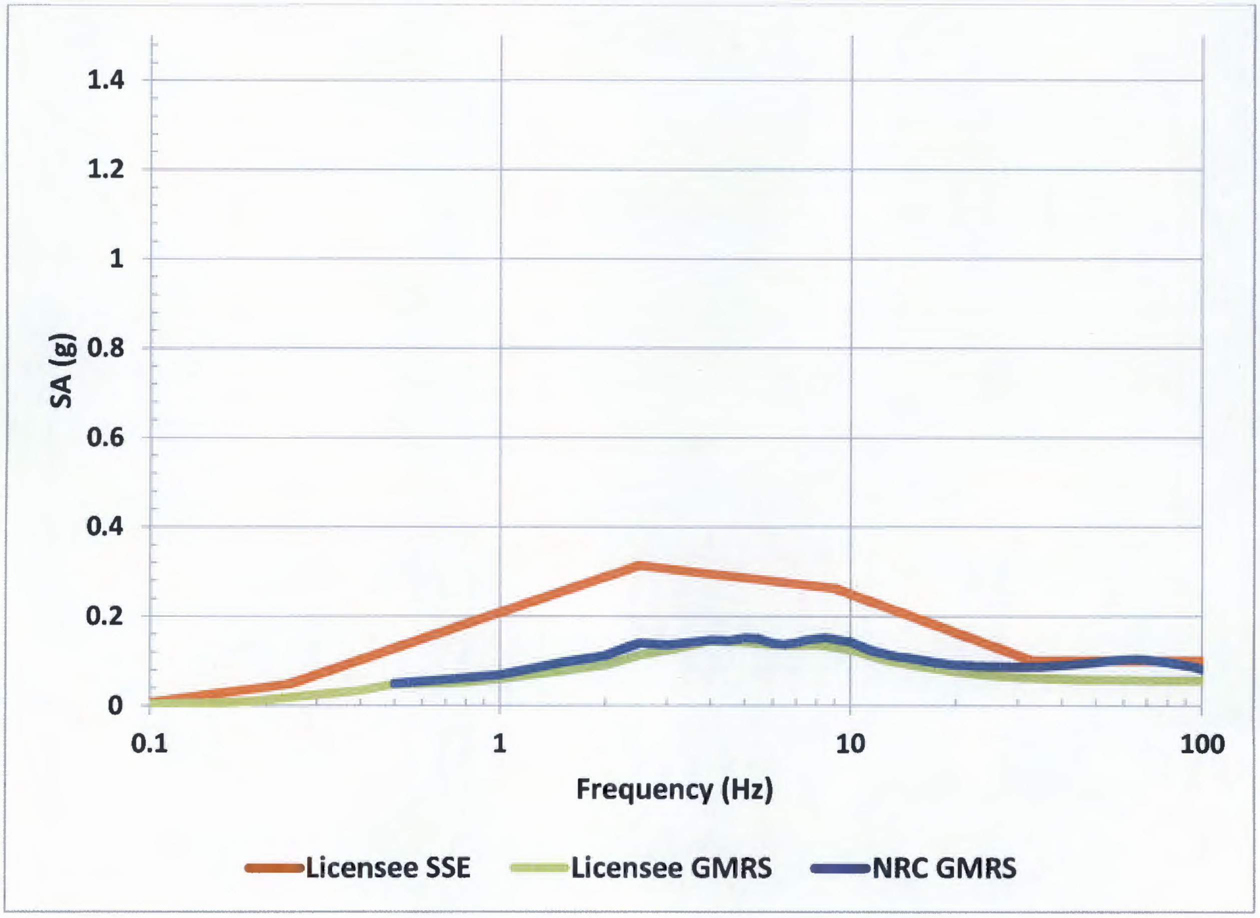

1.E-01
1.E-02
1.E-03
1.E-04
1.E-05
1.E-06
1.E-07
1.E-08
Mean Annual Frequency of Exceedance
0.001
0.01
0.1
1
10
Spectral Acceleration (g)
Licensee 1 Hz
Licensee 10 Hz
Licensee PGA
NRC 1 Hz
NRC 10 Hz
NRC PGA

Figure 3.4-1 Comparison of the Staff's GMRS with Licensee's GMRS and the SSE for the South Texas Project site



If you have any questions, please contact me at (301) 415-1617 or at Frankie.Vega@nrc.gov.

Sincerely,

/RA/

Frankie Vega, Project Manager
Hazards Management Branch
Japan Lessons-Learned Division
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure:
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Hazard Evaluation and Screening Report

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