L44 140331 004

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-14-038

March 31, 2014

10 CFR 50.4
10 CFR 50.54(f)

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Bellefonte Nuclear Plant, Units 1 and 2
Construction Permit Nos. CPPR-122 and CPPR-123
NRC Docket Nos. 50-438 and 50-439

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Sequoyah Nuclear Plant, Units 1 and 2
Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327 and 50-328

Watts Bar Nuclear Plant, Unit 1
Facility Operating License No. NPF-90
NRC Docket No. 50-390

Watts Bar Nuclear Plant, Unit 2
Construction Permit No. CPPR-92
NRC Docket No. 50-391

Subject: **Tennessee Valley Authority's Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident**

References:

1. NRC Letter, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 12, 2012 (ML12056A046)

2. NEI Letter, “Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations,” dated April 9, 2013 (ML13101A379)

3. NRC Letter, “Electric Power Research Institute Final Draft Report XXXXXX, “Seismic Evaluation Guidance: Augmented Approach for the Resolution of Near-Term Task Force Recommendation 2.1: Seismic,” as an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluations,” dated May 7, 2013 (ML13106A331)

4. EPRI Report 1025287, “Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” November 2012 (ML12333A170)

5. NRC Letter, “Endorsement of Electric Power Research Institute Final Draft Report 1025287, “Seismic Evaluation Guidance,”” dated February 15, 2013 (ML12319A074)

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued Reference 1 to all power reactor licensees and holders of construction permits in active or deferred status. Enclosure 1 of Reference 1 requested each addressee located in the Central and Eastern United States (CEUS) to submit a Seismic Hazard Evaluation and Screening Report within 1.5 years from the date of Reference 1.

In Reference 2, the Nuclear Energy Institute (NEI) requested NRC agreement to delay submittal of the final CEUS Seismic Hazard Evaluation and Screening Reports so that an update to the Electric Power Research Institute (EPRI) ground motion attenuation model could be completed and used to develop that information. NEI proposed that descriptions of subsurface materials and properties and base case velocity profiles be submitted to the NRC by September 12, 2013, with the remaining seismic hazard and screening information submitted by March 31, 2014. In Reference 3, the NRC agreed with NEI’s proposal.

Reference 4 contains industry guidance and detailed information to be included in the Seismic Hazard Evaluation and Screening Report submittals. The NRC endorsed this industry guidance in Reference 5.

The enclosed Seismic Hazard Evaluation and Screening Reports provide the information described in Section 4 of Reference 4 in accordance with the schedule identified in Reference 2. Specifically, Enclosures 1, 2, 3, and 4 provide the reports for the Bellefonte, Browns Ferry, Sequoyah, and Watts Bar Nuclear Plants, respectively.

This letter contains no new regulatory commitments.

Should you have any questions concerning the content of this letter, please contact Kevin Casey at (423) 751-8523.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 31st day of March 2014.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:
1. Seismic Hazard and Screening Report for Tennessee Valley Authority's Bellefonte Nuclear Plant
2. Seismic Hazard and Screening Report for Tennessee Valley Authority's Browns Ferry Nuclear Plant
3. Seismic Hazard and Screening Report for Tennessee Valley Authority's Sequoyah Nuclear Plant
4. Seismic Hazard and Screening Report for Tennessee Valley Authority's Watts Bar Nuclear Plant

cc (Enclosures):

NRR Director - NRC Headquarters
NRO Director - NRC Headquarters
NRC Regional Administrator - Region II
NRR Project Manager - Bellefonte Nuclear Plant
NRR Project Manager - Browns Ferry Nuclear Plant
NRR Project Manager - Sequoyah Nuclear Plant
NRR Project Manager - Watts Bar Nuclear Plant, Unit 1
NRR Project Manager - Watts Bar Nuclear Plant, Unit 2
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
NRC Senior Resident Inspector - Sequoyah Nuclear Plant
NRC Senior Resident Inspector - Watts Bar Nuclear Plant, Unit 1
NRC Senior Resident Inspector - Watts Bar Nuclear Plant, Unit 2

ENCLOSURE 1

SEISMIC HAZARD AND SCREENING REPORT FOR TENNESSEE VALLEY AUTHORITY'S BELLEFONTE NUCLEAR PLANT

Seismic Hazard and Screening Report for Bellefonte Nuclear Plant

Table of Contents

1.0 Introduction

Following the accident at the Fukushima Daiichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the NRC Commission established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter that requests information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements. Depending on the comparison between the reevaluated seismic hazard and the current design basis, the result is either no further risk evaluation or the performance of a seismic risk assessment. Risk assessment approaches acceptable to the staff include a seismic probabilistic risk assessment (SPRA), or a seismic margin assessment (SMA). Based upon this information, the NRC staff will determine whether additional regulatory actions are necessary.

This report provides the information requested in items (1) through (7) of the “Requested Information” section and Attachment 1 of the 50.54(f) letter pertaining to NTTF Recommendation 2.1 for Bellefonte Nuclear Plant , located in Jackson County, Alabama. In providing this information, Tennessee Valley Authority followed the guidance provided in the Seismic Evaluation Guidance: Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic (EPRI, 2013a).

The original geologic and seismic siting investigations for Bellefonte Nuclear Plant were performed in accordance with Appendix A to 10 CFR Part 100 and meet General Design Criterion 2 in Appendix A to 10 CFR Part 50. The Safe Shutdown Earthquake Ground Motion (SSE) was developed in accordance with Appendix A to 10 CFR Part 100 and used for the design of seismic Category I systems, structures and components.

In response to the 50.54(f) letter and following the guidance provided in the SPID (EPRI, 2013a), a seismic hazard reevaluation was performed. For screening purposes, a Ground Motion Response Spectrum (GMRS) was developed. Based on the results of the screening evaluation, Bellefonte Nuclear Plant screens-in for a risk evaluation, a Spent Fuel Pool evaluation and a High Frequency Confirmation.

2.0 Seismic Hazard Reevaluation

Bellefonte Nuclear Plant is located approximately 38 miles east of Huntsville, Alabama, on the west bank of Guntersville Reservoir at river mile 391.5 (TVA, Amendment 30, Section 2.1.1). Bellefonte Nuclear Plant is located in the Browns Valley-Sequatchie Valley segment of the Cumberland Plateau section of the Appalachian Plateaus province. This section in the southwestern most of the seven sections comprising the Appalachian Plateaus proving and extends from New York to the Coastal Plain in northwestern Alabama. It is bounded on the west by the Coastal Plain province and Interior Low Plateaus province and on the east by the Valley and Ridge province. (TVA, Amendment 30, Section 2.5).

The evaluation of the earthquake hazard at the Bellefonte Nuclear Plant site involves a consideration of the known seismic history of a large surrounding area. The largest historic earthquake known in the Southern Appalachian Tectonic Province is the 1897 Giles County, Virginia earthquake. The SSE for the plant has been established as having a maximum top of rock horizontal and vertical peak ground acceleration (PGA) of 0.18g. The most seismically active areas are described in the following summary. (TVA, Amendment 30, Sections 2.5.2.1 and 2.5.2.4).

a. The Upper Mississippi Valley, especially the New Madrid region of Arkansas, Kentucky, Missouri, and Tennessee. A few major earthquakes and thousands of light to moderately strong shocks have been centered in the Upper Mississippi Valley. Light to moderate shocks are still occurring at a frequency of a few per year in this zone. This region is more than 250 miles northwest of the Bellefonte Nuclear Plant site.
b. The Lower Wabash Valley of Illinois and Indiana. This area has been the center of several moderately strong earthquakes. The area is approximately 330 miles northwest of Bellefonte Nuclear Plant.
c. Charleston area, South Carolina. One of the country's greatest earthquakes occurred near Charleston in 1886. Earlier, many light to moderate shocks had been centered in the area long before the major earthquake. Charleston is 285 miles east of the Bellefonte Nuclear Plant site.
d. The Appalachian Mountains of Eastern Tennessee and Western North Carolina. The mountain belt of eastern Tennessee and western North Carolina is a region of continuing minor activity. Light to moderate shocks occur at an average frequency of one or two per year. The activity is not uniform, as periods of several shocks per year are followed by longer periods of no perceptible shocks. This region is centered more than 100 miles to the east of the Bellefonte Nuclear Plant site.

2.1 Regional and Local Geology

The Bellefonte Nuclear Plant is located in the Browns Valley-Sequatchie Valley segment of the Cumberland Plateau section of the Appalachian Plateaus province of the Appalachian Highlands. The Appalachian Plateaus province is bordered by the Valley and Ridge province on the east, the Interior Low Plateaus to the northwest, and the Coastal Plain to the southwest. It extends from northwestern New York to northwestern Alabama. From its maximum width of more than 200 miles, it begins to narrow in eastern Kentucky until it is barely 30 miles wide in Tennessee. The width in Alabama is 50 miles. This province is essentially a broad syncline in rocks of Late Paleozoic age, bounded on all sides by escarpment that reflect the regional synclinal structure. The province is underlain by Paleozoic sedimentary rocks which are basically flat-lying (TVA, Amendment 30, Section 2.5.1.1.1 and 2.5.1.1.3).

On the east the Appalachian Plateaus province is bounded by the Valley and Ridge province, which is made up of a series of folded and faulted ridges and valleys. Thirty to forty thousand feet of Paleozoic sedimentary rocks were involved in the folding and faulting of the Valley and Ridge province (TVA, Amendment 30, Section 2.5.1.1.3).

East of the Valley and Ridge is the Blue Ridge province, which is underlain by Lower Cambrian and Upper Precambrian sedimentary rocks and Precambrian basement complex plutonic and gneissic rocks (TVA, Amendment 30, Section 2.5.1.1.3).

Farther to the east, beyond the Blue Ridge Province, is the Piedmont province, whose rocks are mainly metamorphic and plutonic. The degree of metamorphism increases eastward from the Valley and Ridge into the Piedmont province and reaches a maximum at the sillimanite zone which borders the Carolina slate belt.

West of the Appalachian Plateaus province is the Interior Low Plateaus Province of the Interior Plains. This province contains relatively flat-lying rocks ranging in age from Ordovician to Cretaceous.

The Bellefonte reservation is located near the cities of Hollywood and Scottsboro in Jackson County, northeast Alabama. The reservation is on the right bank of Guntersville Reservoir at river mile 391.5. The main plant facilities are separated from the reservoir by River Ridge, a low line of hills rising about 200 feet above water level. The rock supported structures are founded upon limestone and interbedded shale of the Chickamauga Formation of Middle Ordovician age (TVA, Amendment 30, Section 2.5).

2.2 Probabilistic Seismic Hazard Analysis

2.2.1 Probabilistic Seismic Hazard Analysis Results

In accordance with the 50.54(f) letter and following the guidance in the SPID (EPRI, 2013a), a probabilistic seismic hazard analysis (PSHA) was completed using the recently developed Central and Eastern United States Seismic Source Characterization (CEUS-SSC) for Nuclear Facilities (CEUS-SSC, 2012) together with the updated EPRI Ground-Motion Model (GMM) for the CEUS (EPRI, 2013b). For the PSHA, a minimum moment magnitude cutoff of 5.0 was used, as specified in the 50.54(f) letter (U.S. NRC, 2012) (EPRI, 2013c).

For the PSHA, the CEUS-SSC background seismic source zones out to a distance of 400 miles (640 km) around Bellefonte were included. This distance exceeds the 200 mile (320 km) recommendation contained in (U. S. NRC, 2007) and was chosen for completeness. For each of the CEUS-SSC sources, the mid-continent version of the updated CEUS EPRI GMM was used (EPRI, 2013c).

2.2.2 Base Rock Seismic Hazard Curves

Bellefonte is a hard-rock site. Consistent with the SPID (EPRI,2013a), hard-rock seismic hazard curves are shown below in Figure 2.3.7-1 at the SSE control point elevation (EPRI, 2013c).

2.3 Site Response Evaluation

Based on information describing the Bellefonte Nuclear Plant site presented in Section 2.3.1, the geologic layers underlying the foundation of the plant consist of hard rock (Vs $\geq$ 9280 fps). Therefore no site-specific evaluation of site amplification was performed for Bellefonte (EPRI, 2013c).

2.3.1 Description of Subsurface Material

Bellefonte Nuclear Plant is located on a gently dipping (i.e., about 15° to 20°) southeast limb of the Sequatchie anticline in a long anticlinal valley (Browns Valley) in the dissected Cumberland Plateau section within the Appalachian Plateau Physiographic Province. The TVA Bellefonte property is located on the right bank of Guntersville Reservoir on the Tennessee River at river mile 391.5 in Jackson County, Alabama (EPRI, 2013c).

The bedrock at the Bellefonte site consists of alternating layers of gently dipping Ordovician limestone (originally mapped as the Chickamauga Limestone) of the Stones River Group, the Nashville Group, and the Sequatchie Formation. The Stones River Group is comprised of three subunits (Upper Stones River, Middle Stones River, and Lower Stones River) that differ slightly from one another in composition and texture, containing alternating beds of limestone to dolomitic limestone and argillaceous and silty limestone, with some cherty limestone. The

Middle Stones River (MSR) is further divided into six distinct lithologic units, designated units A through F. These subunits of the MSR comprise a total thickness of about 453 feet within the 1050 feet thick Stones River Group (EPRI, 2013c).

Data reported in Bechtel (2012) indicate that rock supporting reactor structures has shear-wave velocities of 9,280 fps or greater. Therefore the Bellefonte site is treated as a hard-rock site (EPRI, 2013c).

2.3.2 Development of Base Case Profiles and Nonlinear Material Properties

Sections 2.3.2—2.3.6 are not needed because Bellefonte is a hard rock site (EPRI, 2013c).

2.3.7 Control Point Seismic Hazard Curves

The procedure to develop probabilistic seismic hazard curves for hard rock follows standard techniques documented in the technical literature (e.g., McGuire, 2004). Separate seismic hazard calculations are conducted for the 7 spectral frequencies for which ground motion equations are available (100 Hz=peak ground acceleration or PGA, 25 Hz, 10 Hz, 5 Hz, 2.5 Hz, 1 Hz, and 0.5 Hz). As discussed in Section 2.2.1, ground motion equations from the updated EPRI Ground-Motion Model (GMM) for the CEUS (EPRI, 2013b) were used for the calculation of rock hazard. All spectra accelerations presented herein correspond to 5% of critical damping. Figure 2.3.7-1 shows the mean hard-rock seismic hazard curves for the 7 spectral frequencies. The digital values for the mean and fractile hazard curves are provided in Appendix A (EPRI, 2013c).

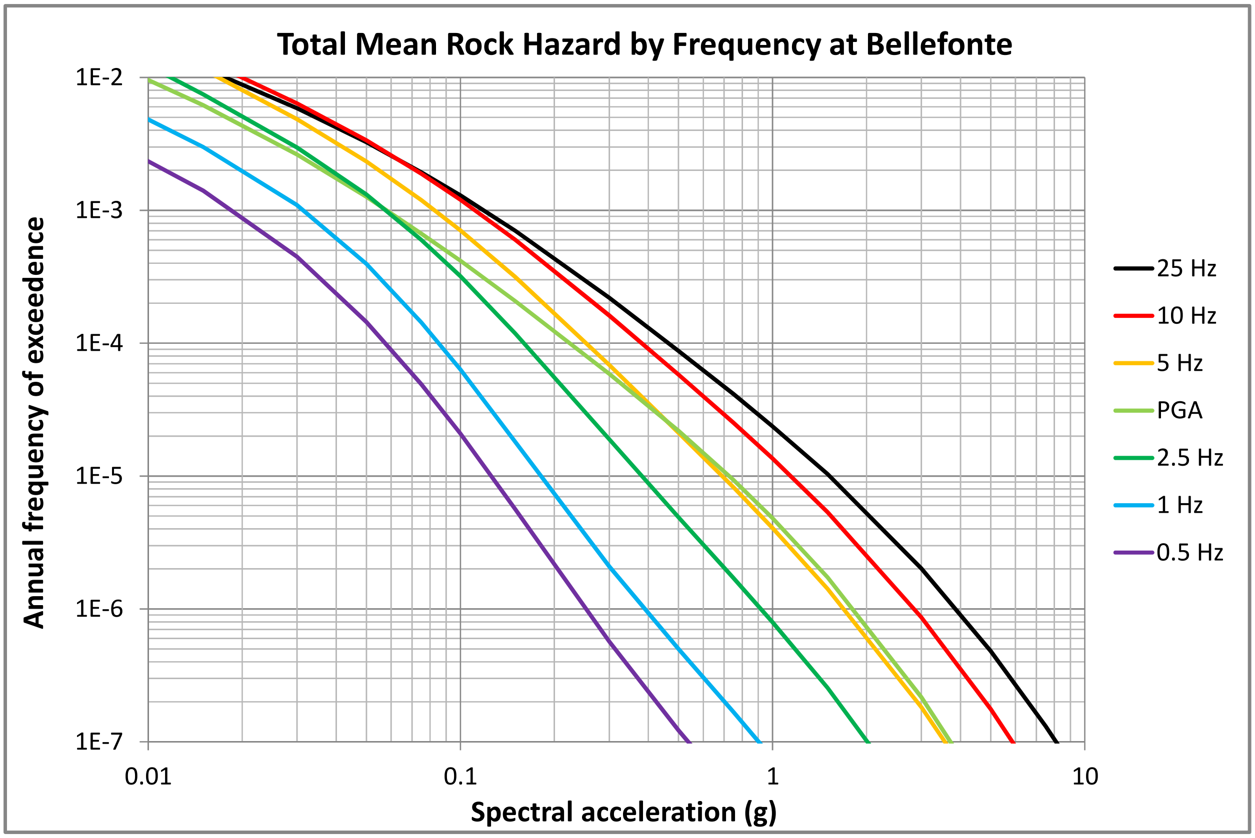


Figure 2.3.7-1. Control point mean hazard curves for oscillator spectral frequencies of 0.5, 1, 2.5, 5, 10, 25 and 100 Hz at Bellefonte (EPRI, 2013c).

2.4 Control Point Response Spectra

The control point hazard curves described above have been used to develop Uniform Hazard Response Spectra (UHRS) and the GMRS. The UHRS were obtained through linear interpolation in log-log space to estimate the spectral acceleration at each spectral frequency for the 10^{-4} and 10^{-5} per year hazard levels.

The 10^{-4} and 10^{-5} UHRS, along with a design factor are used to compute the GMRS at the control point using the criteria in Regulatory Guide 1.208 (U.S. NRC, 2007). Table 2.4-1 shows the UHRS and GMRS accelerations for a range of frequencies. (EPRI, 2013c)

Table 2.4-1. UHRS and GMRS for Bellefonte (EPRI, 2013c)

Freq. (Hz)	10^{-4} UHRS (g)	10^{-5} UHRS (g)	GMRS
100	2.24E-01	7.26E-01	3.44E-01
90	2.42E-01	7.85E-01	3.72E-01
80	2.74E-01	8.89E-01	4.22E-01
70	3.22E-01	1.05E+00	4.97E-01
60	3.82E-01	1.24E+00	5.90E-01
50	4.37E-01	1.42E+00	6.75E-01
45	4.57E-01	1.49E+00	7.06E-01
40	4.70E-01	1.53E+00	7.26E-01
35	4.76E-01	1.55E+00	7.36E-01
30	4.74E-01	1.55E+00	7.34E-01
25	4.65E-01	1.52E+00	7.20E-01
20	4.58E-01	1.47E+00	6.98E-01
15	4.35E-01	1.36E+00	6.48E-01
12.5	4.13E-01	1.27E+00	6.07E-01
10	3.81E-01	1.14E+00	5.50E-01
9	3.60E-01	1.07E+00	5.15E-01
8	3.37E-01	9.83E-01	4.76E-01
7	3.12E-01	8.93E-01	4.34E-01
6	2.84E-01	7.95E-01	3.88E-01
5	2.53E-01	6.90E-01	3.38E-01
4	2.21E-01	5.78E-01	2.86E-01
3	1.83E-01	4.53E-01	2.26E-01
2.5	1.60E-01	3.82E-01	1.92E-01
2	1.46E-01	3.36E-01	1.71E-01
1.5	1.22E-01	2.70E-01	1.38E-01
1.25	1.05E-01	2.29E-01	1.17E-01
1	8.52E-02	1.82E-01	9.36E-02
0.9	8.26E-02	1.77E-01	9.13E-02
0.8	7.87E-02	1.70E-01	8.73E-02
0.7	7.32E-02	1.59E-01	8.15E-02
0.6	6.61E-02	1.44E-01	7.39E-02
0.5	5.74E-02	1.26E-01	6.44E-02
0.4	4.59E-02	1.00E-01	5.16E-02
0.3	3.45E-02	7.54E-02	3.87E-02
0.2	2.30E-02	5.02E-02	2.58E-02
0.167	1.92E-02	4.20E-02	2.15E-02
0.125	1.44E-02	3.14E-02	1.61E-02
0.1	1.15E-02	2.51E-02	1.29E-02

The 10^{-4} and 10^{-5} UHRS and GMRS are plotted in Figure 2.4-1.

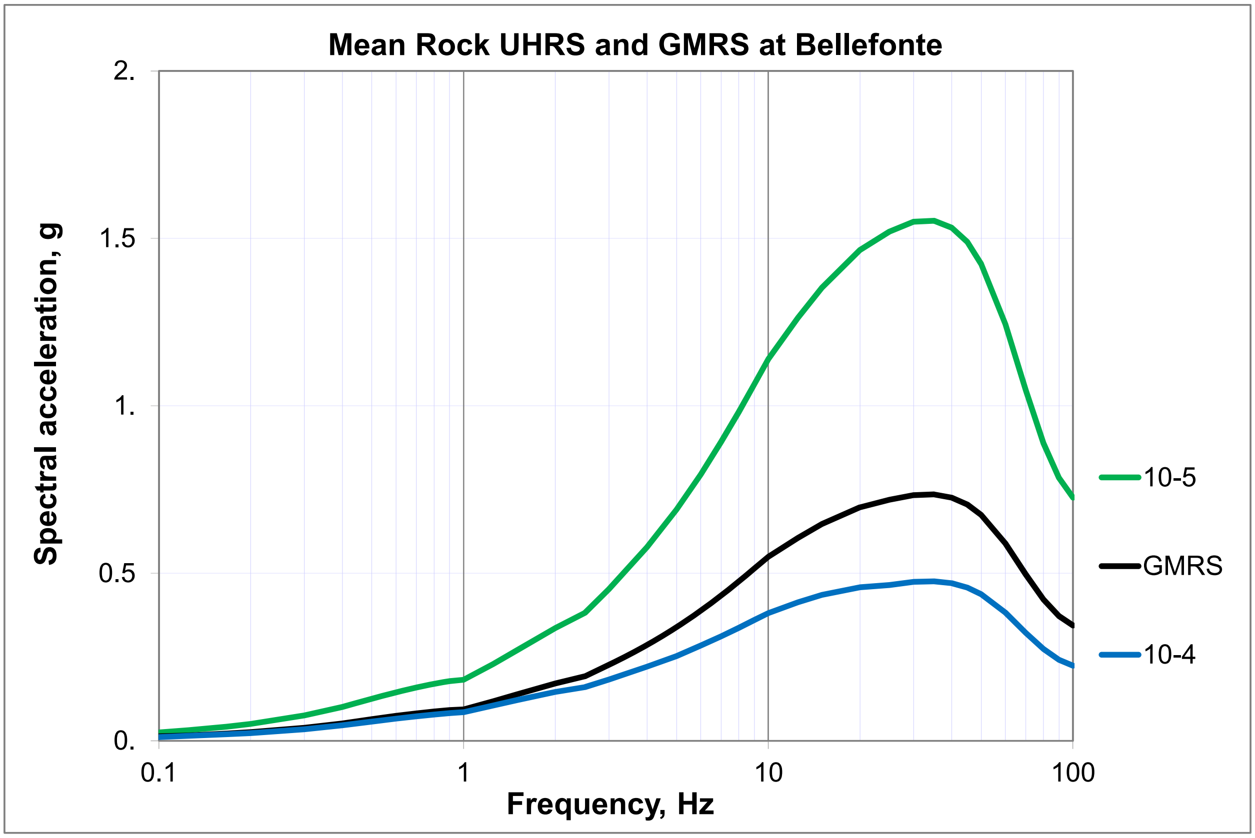


Figure 2.4-1. 10^{-4} and 10^{-5} UHRS and GMRS for Bellefonte Nuclear Plant (EPRI, 2013c).

3.0 Safe Shutdown Earthquake Ground Motion

The design basis for Bellefonte Nuclear Plant is identified in the Final Safely Analysis Report (TVA, Amendment 30).

3.1 SSE Description of Spectral Shape

The maximum acceleration of 0.18 g was used in conjunction with Regulatory Guide 1.60 Revision 1 to define the response spectra for the SSE. Both the horizontal and vertical response spectra are anchored to the same PGA in accordance with Regulatory Guide 1.60 Revision 1 (TVA, Amendment 30, Section 2.5.1.6).

Table 3.1-1 shows the Spectral Acceleration (SA) values as a function of the frequency for the 5% damped horizontal SSE.

Table 3.1-1. SSE for Bellefonte Nuclear Plant (TVA, Amendment 30)

Freq. (Hz)	1	2.5	5	9	33	100
SA (g)	0.26	0.56	.51	0.47	0.18	0.18

3.2 Control Point Elevation

The SSE control point is defined at the top of hard rock at the reactor building foundation, elevation 612 feet.

4.0 Screening Evaluation

In accordance with SPID (EPRI, 2013a) Section 3, a screening evaluation was performed as described below.

4.1 Risk Evaluation Screening (1 to 10 Hz)

In the 1 to 10 Hz part of the response spectrum, the GMRS exceeds the SSE. Therefore, Bellefonte Nuclear Plant screens in for a risk evaluation.

4.2 High Frequency Screening (>10 Hz)

For the range above 10 Hz, the GMRS exceeds the SSE. Therefore, Bellefonte Nuclear Plant screens in for a high frequency confirmation.

4.3 Spent Fuel Pool Evaluation Screening (1 to 10 Hz)

In the 1 to 10 Hz part of the response spectrum, the GMRS exceeds the SSE. Therefore, Bellefonte Nuclear Plant screens in for a spent fuel pool evaluation.

5.0 Interim Actions

Bellefonte Nuclear Plant remains in a "deferred plant" construction permit status. Consequently, no interim actions are planned at this time.

6.0 Conclusions

In accordance with the 50.54(f) request for information, a seismic hazard and screening evaluation was performed for Bellefonte Nuclear Plant. A GMRS was developed solely for purpose of screening for additional evaluations in accordance with the SPID. Based on the results of the screening evaluation, the Bellefonte Nuclear Plant screen in for risk evaluation, a Spent Fuel Pool evaluation, and High Frequency confirmation.

7.0 References

Bechtel (2012). Dynamic Soil Column for Unit 1 Reactor Building, Letter 25644-012-TCM-GAM-00004 from Moreschi to Matyas dated October 2, 2012.

CEUS-SSC (2012). “Central and Eastern United States Seismic Source Characterization for Nuclear Facilities”, U.S. Nuclear Regulatory Commission Report, NUREG-2115; EPRI Report 1021097, 6 Volumes; DOE Report# DOE/NE-0140.

EPRI (2013a). Seismic Evaluation Guidance Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, Elec. Power Research Institute Report 1025287, February 2013.

EPRI (2013b). “EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project”, Electric Power Research Institute, Palo Alto, CA, Report 3002000717, June, 2 volumes.

EPRI (2013c). “Bellefonte Seismic Hazard and Screening Report,” Electric Power Research Institute, Palo Alto, CA, dated October 30, 2013.

McGuire, R.K. (2004). *Seismic Hazard and Risk Analysis*, Earthquake Eng. Res. Inst., Monograph MNO-10.

TVA (Amendment 30). “Bellefonte Nuclear Plant Final Safety Analysis Report,” Amendment 30.

U.S. NRC (2007). “A performance-based approach to define the site-specific earthquake ground motion,” U.S. Nuclear Regulatory Commission Reg. Guide 1.208.

U.S. NRC (2012) NRC (E Leeds and M Johnson) Letter to All Power Reactor Licensees et al., “Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident,” March 12, 2012.

Appendix A

Tabulated Data

Table A-1a. Mean and Fractile Seismic Hazard Curves for PGA at Bellefonte Nuclear Power Plant Units 1 and 2 (EPRI, 2013c)

AMPS(g)	MEAN	0.05	0.16	0.5	0.84	0.95
0.0005	9.74E-02	4.70E-02	8.23E-02	9.93E-02	9.93E-02	9.93E-02
0.001	7.11E-02	2.88E-02	5.75E-02	6.93E-02	8.98E-02	9.93E-02
0.005	1.92E-02	8.12E-03	1.23E-02	1.79E-02	2.35E-02	4.43E-02
0.01	9.59E-03	3.68E-03	5.35E-03	8.47E-03	1.21E-02	2.49E-02
0.015	6.17E-03	2.04E-03	2.96E-03	5.20E-03	8.35E-03	1.69E-02
0.03	2.63E-03	6.00E-04	8.98E-04	1.87E-03	4.07E-03	8.12E-03
0.05	1.26E-03	2.35E-04	3.57E-04	7.55E-04	1.95E-03	4.50E-03
0.075	6.70E-04	1.16E-04	1.82E-04	3.73E-04	9.79E-04	2.57E-03
0.1	4.18E-04	7.23E-05	1.15E-04	2.32E-04	5.75E-04	1.60E-03
0.15	2.07E-04	3.79E-05	6.17E-05	1.20E-04	2.80E-04	7.66E-04
0.3	5.87E-05	1.08E-05	1.87E-05	3.95E-05	8.35E-05	1.92E-04
0.5	2.19E-05	3.63E-06	6.64E-06	1.55E-05	3.28E-05	6.64E-05
0.75	9.35E-06	1.31E-06	2.60E-06	6.54E-06	1.46E-05	2.84E-05
1	4.83E-06	5.66E-07	1.18E-06	3.33E-06	7.66E-06	1.51E-05
1.5	1.73E-06	1.44E-07	3.33E-07	1.11E-06	2.80E-06	5.83E-06
3	2.17E-07	8.00E-09	2.25E-08	1.10E-07	3.37E-07	8.98E-07
5	3.43E-08	6.73E-10	1.95E-09	1.31E-08	4.90E-08	1.69E-07
7.5	6.35E-09	1.90E-10	3.19E-10	1.84E-09	8.12E-09	3.52E-08
10	1.69E-09	1.34E-10	1.77E-10	4.83E-10	2.16E-09	9.93E-09

Table A-1b. Mean and Fractile Seismic Hazard Curves for 25 Hz at Bellefonte Nuclear Power Plant Units 1 and 2 (EPRI, 2013c)

AMPS(g)	MEAN	0.05	0.16	0.5	0.84	0.95
0.0005	1.09E-01	6.73E-02	9.65E-02	9.93E-02	9.93E-02	9.93E-02
0.001	8.76E-02	4.50E-02	7.45E-02	8.85E-02	9.93E-02	9.93E-02
0.005	3.10E-02	1.42E-02	2.16E-02	2.92E-02	3.68E-02	6.36E-02
0.01	1.70E-02	7.55E-03	1.08E-02	1.55E-02	2.07E-02	4.01E-02
0.015	1.18E-02	4.98E-03	7.03E-03	1.05E-02	1.49E-02	2.88E-02
0.03	5.86E-03	2.04E-03	2.92E-03	4.98E-03	8.23E-03	1.46E-02
0.05	3.26E-03	9.51E-04	1.34E-03	2.53E-03	4.98E-03	8.72E-03
0.075	1.93E-03	4.83E-04	7.03E-04	1.38E-03	3.05E-03	5.66E-03
0.1	1.29E-03	2.96E-04	4.37E-04	8.72E-04	2.01E-03	4.07E-03
0.15	7.01E-04	1.49E-04	2.32E-04	4.56E-04	1.04E-03	2.29E-03
0.3	2.19E-04	4.98E-05	8.00E-05	1.51E-04	3.05E-04	6.54E-04
0.5	8.76E-05	2.16E-05	3.52E-05	6.73E-05	1.25E-04	2.35E-04
0.75	4.13E-05	1.01E-05	1.67E-05	3.37E-05	6.26E-05	1.04E-04
1	2.38E-05	5.42E-06	9.37E-06	1.98E-05	3.68E-05	5.75E-05
1.5	1.04E-05	2.10E-06	3.84E-06	8.60E-06	1.64E-05	2.53E-05
3	2.02E-06	2.92E-07	5.83E-07	1.60E-06	3.33E-06	5.58E-06
5	4.83E-07	4.63E-08	1.04E-07	3.37E-07	8.12E-07	1.53E-06
7.5	1.31E-07	8.47E-09	2.07E-08	8.00E-08	2.22E-07	4.70E-07
10	4.73E-08	2.22E-09	5.91E-09	2.53E-08	8.00E-08	1.82E-07

Table A-1c. Mean and Fractile Seismic Hazard Curves for 10 Hz at Bellefonte Nuclear Power Plant Units 1 and 2 (EPRI, 2013c)

AMPS(g)	MEAN	0.05	0.16	0.5	0.84	0.95
0.0005	1.17E-01	9.37E-02	9.93E-02	9.93E-02	9.93E-02	9.93E-02
0.001	1.00E-01	6.93E-02	8.72E-02	9.93E-02	9.93E-02	9.93E-02
0.005	3.91E-02	2.01E-02	2.84E-02	3.90E-02	4.83E-02	6.26E-02
0.01	2.06E-02	1.02E-02	1.38E-02	2.01E-02	2.57E-02	3.79E-02
0.015	1.37E-02	6.54E-03	8.85E-03	1.31E-02	1.74E-02	2.64E-02
0.03	6.38E-03	2.57E-03	3.57E-03	5.83E-03	8.85E-03	1.31E-02
0.05	3.36E-03	1.10E-03	1.57E-03	2.84E-03	5.12E-03	7.66E-03
0.075	1.89E-03	5.12E-04	7.55E-04	1.44E-03	3.01E-03	4.90E-03
0.1	1.20E-03	2.92E-04	4.37E-04	8.60E-04	1.90E-03	3.33E-03
0.15	5.99E-04	1.32E-04	2.04E-04	4.07E-04	9.11E-04	1.77E-03
0.3	1.61E-04	3.52E-05	5.83E-05	1.15E-04	2.32E-04	4.50E-04
0.5	5.79E-05	1.32E-05	2.22E-05	4.50E-05	8.60E-05	1.53E-04
0.75	2.51E-05	5.42E-06	9.51E-06	2.01E-05	3.90E-05	6.45E-05
1	1.36E-05	2.68E-06	4.90E-06	1.10E-05	2.16E-05	3.47E-05
1.5	5.36E-06	9.24E-07	1.74E-06	4.25E-06	8.72E-06	1.40E-05
3	8.65E-07	9.79E-08	2.07E-07	6.26E-07	1.44E-06	2.64E-06
5	1.76E-07	1.29E-08	3.01E-08	1.10E-07	2.96E-07	6.17E-07
7.5	4.18E-08	2.01E-09	5.05E-09	2.19E-08	7.03E-08	1.64E-07
10	1.36E-08	5.58E-10	1.34E-09	6.17E-09	2.29E-08	5.66E-08

Table A-1d. Mean and Fractile Seismic Hazard Curves for 5 Hz at Bellefonte Nuclear Power Plant Units 1 and 2 (EPRI, 2013c)

AMPS(g)	MEAN	0.05	0.16	0.5	0.84	0.95
0.0005	1.18E-01	9.37E-02	9.93E-02	9.93E-02	9.93E-02	9.93E-02
0.001	1.00E-01	6.73E-02	8.35E-02	9.93E-02	9.93E-02	9.93E-02
0.005	3.71E-02	1.77E-02	2.64E-02	3.52E-02	4.90E-02	5.66E-02
0.01	1.83E-02	8.60E-03	1.23E-02	1.77E-02	2.46E-02	2.92E-02
0.015	1.15E-02	5.35E-03	7.45E-03	1.10E-02	1.57E-02	1.90E-02
0.03	4.84E-03	1.87E-03	2.72E-03	4.50E-03	7.03E-03	8.98E-03
0.05	2.33E-03	7.03E-04	1.05E-03	1.98E-03	3.63E-03	5.20E-03
0.075	1.19E-03	2.96E-04	4.56E-04	8.98E-04	1.90E-03	3.14E-03
0.1	7.06E-04	1.60E-04	2.49E-04	4.90E-04	1.08E-03	2.04E-03
0.15	3.14E-04	6.54E-05	1.05E-04	2.07E-04	4.56E-04	9.51E-04
0.3	6.86E-05	1.46E-05	2.42E-05	4.83E-05	9.93E-05	1.90E-04
0.5	2.13E-05	4.37E-06	7.55E-06	1.64E-05	3.28E-05	5.66E-05
0.75	8.21E-06	1.51E-06	2.76E-06	6.45E-06	1.34E-05	2.19E-05
1	4.07E-06	6.45E-07	1.25E-06	3.14E-06	6.73E-06	1.10E-05
1.5	1.42E-06	1.72E-07	3.63E-07	1.02E-06	2.42E-06	4.13E-06
3	1.82E-07	1.13E-08	2.84E-08	1.05E-07	3.14E-07	6.26E-07
5	3.09E-08	1.11E-09	3.01E-09	1.40E-08	5.20E-08	1.20E-07
7.5	6.36E-09	2.46E-10	4.90E-10	2.25E-09	1.01E-08	2.68E-08
10	1.88E-09	1.72E-10	2.10E-10	6.26E-10	2.84E-09	8.35E-09

Table A-1e. Mean and Fractile Seismic Hazard Curves for 2.5 Hz at Bellefonte Nuclear Power Plant Units 1 and 2 (EPRI, 2013c)

AMPS(g)	MEAN	0.05	0.16	0.5	0.84	0.95
0.0005	1.11E-01	8.47E-02	9.51E-02	9.93E-02	9.93E-02	9.93E-02
0.001	8.84E-02	5.66E-02	6.93E-02	8.72E-02	9.93E-02	9.93E-02
0.005	2.66E-02	1.31E-02	1.82E-02	2.49E-02	3.57E-02	4.31E-02
0.01	1.22E-02	5.91E-03	8.00E-03	1.18E-02	1.64E-02	2.04E-02
0.015	7.44E-03	3.37E-03	4.63E-03	7.03E-03	1.04E-02	1.27E-02
0.03	2.97E-03	9.51E-04	1.44E-03	2.64E-03	4.50E-03	6.09E-03
0.05	1.31E-03	2.96E-04	4.77E-04	1.01E-03	2.16E-03	3.37E-03
0.075	6.01E-04	1.07E-04	1.77E-04	3.90E-04	9.79E-04	1.87E-03
0.1	3.19E-04	4.98E-05	8.35E-05	1.90E-04	4.98E-04	1.08E-03
0.15	1.18E-04	1.69E-05	2.92E-05	6.45E-05	1.82E-04	4.01E-04
0.3	1.89E-05	2.53E-06	4.70E-06	1.11E-05	3.01E-05	5.42E-05
0.5	4.91E-06	5.35E-07	1.13E-06	3.19E-06	8.12E-06	1.49E-05
0.75	1.71E-06	1.36E-07	3.28E-07	1.04E-06	2.96E-06	5.58E-06
1	7.95E-07	4.70E-08	1.23E-07	4.43E-07	1.42E-06	2.72E-06
1.5	2.55E-07	8.85E-09	2.64E-08	1.21E-07	4.50E-07	9.51E-07
3	2.82E-08	4.43E-10	1.34E-09	8.72E-09	4.63E-08	1.21E-07
5	4.27E-09	1.72E-10	2.22E-10	9.51E-10	6.26E-09	1.95E-08
7.5	8.01E-10	1.21E-10	1.69E-10	2.35E-10	1.11E-09	3.73E-09
10	2.22E-10	1.11E-10	1.23E-10	1.72E-10	3.63E-10	1.10E-09

Table A-1f. Mean and Fractile Seismic Hazard Curves for 1.0 Hz at Bellefonte Nuclear Power Plant Units 1 and 2 (EPRI, 2013c)

AMPS(g)	MEAN	0.05	0.16	0.5	0.84	0.95
0.0005	7.12E-02	3.42E-02	5.12E-02	7.34E-02	9.11E-02	9.93E-02
0.001	4.52E-02	1.90E-02	3.05E-02	4.56E-02	5.83E-02	7.13E-02
0.005	1.02E-02	4.19E-03	6.17E-03	9.65E-03	1.42E-02	1.82E-02
0.01	4.83E-03	1.55E-03	2.46E-03	4.43E-03	7.23E-03	9.37E-03
0.015	3.00E-03	7.13E-04	1.27E-03	2.64E-03	4.77E-03	6.45E-03
0.03	1.10E-03	1.38E-04	2.84E-04	7.89E-04	1.92E-03	3.05E-03
0.05	3.94E-04	3.33E-05	7.13E-05	2.29E-04	6.93E-04	1.29E-03
0.075	1.43E-04	9.93E-06	2.13E-05	7.23E-05	2.35E-04	5.35E-04
0.1	6.33E-05	4.07E-06	8.60E-06	2.96E-05	9.93E-05	2.42E-04
0.15	1.81E-05	1.08E-06	2.32E-06	8.12E-06	2.76E-05	6.83E-05
0.3	2.09E-06	9.11E-08	2.39E-07	8.60E-07	3.57E-06	7.77E-06
0.5	4.99E-07	1.21E-08	3.57E-08	1.77E-07	8.12E-07	2.07E-06
0.75	1.67E-07	2.10E-09	7.03E-09	4.63E-08	2.57E-07	7.45E-07
1	7.54E-08	6.26E-10	2.16E-09	1.67E-08	1.10E-07	3.47E-07
1.5	2.28E-08	2.04E-10	4.43E-10	3.47E-09	2.92E-08	1.07E-07
3	2.31E-09	1.21E-10	1.72E-10	2.76E-10	2.10E-09	1.02E-08
5	3.36E-10	1.11E-10	1.21E-10	1.72E-10	3.28E-10	1.36E-09
7.5	6.21E-11	1.11E-10	1.21E-10	1.72E-10	1.72E-10	3.19E-10
10	1.71E-11	1.11E-10	1.15E-10	1.72E-10	1.72E-10	1.87E-10

Table A-1g. Mean and Fractile Seismic Hazard Curves for 0.5 Hz at Bellefonte Nuclear Power Plant Units 1 and 2 (EPRI, 2013c)

AMPS(g)	MEAN	0.05	0.16	0.5	0.84	0.95
0.0005	3.47E-02	1.64E-02	2.60E-02	3.37E-02	4.37E-02	5.20E-02
0.001	2.01E-02	9.37E-03	1.42E-02	1.90E-02	2.60E-02	3.42E-02
0.005	4.79E-03	1.44E-03	2.35E-03	4.37E-03	7.34E-03	9.51E-03
0.01	2.34E-03	3.73E-04	7.55E-04	1.92E-03	3.95E-03	5.66E-03
0.015	1.41E-03	1.38E-04	3.19E-04	1.01E-03	2.57E-03	4.01E-03
0.03	4.47E-04	1.92E-05	5.05E-05	2.19E-04	8.00E-04	1.62E-03
0.05	1.45E-04	3.84E-06	1.04E-05	5.05E-05	2.29E-04	6.17E-04
0.075	4.91E-05	1.01E-06	2.64E-06	1.32E-05	6.83E-05	2.25E-04
0.1	2.09E-05	3.68E-07	9.79E-07	4.83E-06	2.64E-05	9.37E-05
0.15	5.65E-06	8.12E-08	2.25E-07	1.16E-06	7.23E-06	2.42E-05
0.3	5.67E-07	4.31E-09	1.57E-08	9.11E-08	7.45E-07	2.60E-06
0.5	1.22E-07	4.90E-10	1.84E-09	1.38E-08	1.32E-07	6.26E-07
0.75	3.97E-08	1.87E-10	3.79E-10	2.88E-09	3.42E-08	1.98E-07
1	1.79E-08	1.72E-10	2.01E-10	9.37E-10	1.27E-08	8.47E-08
1.5	5.52E-09	1.21E-10	1.72E-10	2.64E-10	2.84E-09	2.35E-08
3	5.92E-10	1.11E-10	1.21E-10	1.72E-10	2.72E-10	1.90E-09
5	9.12E-11	1.11E-10	1.21E-10	1.72E-10	1.72E-10	3.23E-10
7.5	1.77E-11	1.11E-10	1.11E-10	1.72E-10	1.72E-10	1.77E-10
10	5.04E-12	1.11E-10	1.11E-10	1.72E-10	1.72E-10	1.72E-10

ENCLOSURE 2

SEISMIC HAZARD AND SCREENING REPORT FOR TENNESSEE VALLEY AUTHORITY'S BROWNS FERRY NUCLEAR PLANT

Seismic Hazard and Screening Report for Browns Ferry Nuclear Plant

Table of Contents

Page

1.0 Introduction

Following the accident at the Fukushima Daiichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near-Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter (U.S. NRC, 2012) that requests information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter (U.S. NRC, 2012) requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements. Depending on the comparison between the reevaluated seismic hazard and the current design basis, the result is either no further risk evaluation or the performance of a seismic risk assessment. Risk assessment approaches acceptable to the staff include a seismic probabilistic risk assessment (SPRA), or a seismic margin assessment (SMA). Based upon this information, the NRC staff will determine whether additional regulatory actions are necessary.

This report provides the information requested in items (1) through (7) of the "Requested Information" section and Attachment 1 of the 50.54(f) letter (U.S. NRC, 2012) pertaining to NTTF Recommendation 2.1 for the Browns Ferry Nuclear Plant, located in Limestone County, Alabama. In providing this information, the Tennessee Valley Authority followed the guidance provided in the *Seismic Evaluation Guidance: Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic* (EPRI, 2013a). The Augmented Approach, *Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic* (EPRI, 2013c), has been developed as the process for evaluating critical plant equipment prior to performing the complete plant seismic risk evaluations.

Browns Ferry Nuclear Plant located in Limestone County, Alabama was originally licensed for initial power under section 104(b) of the Atomic Energy Act of 1954, as amended, and the regulations of the Atomic Energy Commission set forth in Part 50 of Title 10 of the Code of Federal Regulations (10CFR50). Browns Ferry was not originally licensed under Appendix A of Part 100 – "Seismic and Geologic Siting Criteria for Nuclear Power Plants." However a review of Chapter 2.5 of Browns Ferry Nuclear Plant FSAR reveals that the elements presented in the FSAR meet the general expectations for investigations required to obtain the geologic and seismic data necessary to determine site suitability and provided reasonable assurance that a nuclear power plant can be constructed and operated at the proposed site without undue risk to the health and safety of the public.

In response to the 50.54(f) letter (U.S. NRC, 2012) and following the guidance provided in the SPID (EPRI, 2013a), a seismic hazard reevaluation for Browns Ferry Nuclear Plant was performed. For screening purposes, a Ground Motion Response Spectrum (GMRS) was developed. Based on the results of the screening evaluation, Browns Ferry Nuclear Plant screens-in for a Spent Fuel Pool evaluation and a High Frequency Confirmation. Additionally, based on the results of the screening evaluation, Browns Ferry Nuclear Plant screens-out of a seismic risk evaluation.

2.0 Seismic Hazard Reevaluation

Browns Ferry Nuclear Plant is located approximately 10 miles northwest of the center of Decatur, Alabama, on the north shore of Wheeler Reservoir at Tennessee River mile marker 294 (TVA, Amendment 25.3, Section 2.2.1). The area surrounding Browns Ferry Nuclear Plant lies near the southern margin of the Highland Rim section of the Interior Low Plateaus. This physiographic subdivision is characterized by a young-to-mature plateau of moderate relief. The general level of the ground rises gradually from 600 ft above sea level at the north shore of Wheeler Lake to around 800 ft above sea level at a point 10 miles north in the vicinity of Athens, Alabama. This surface is modified by the drainage patterns of Poplar, Round Island, and Mud Creeks, which flow across it from northeast to southwest. (TVA, Amendment 25.3, Section 2.5.2.4.1)

In order to evaluate the earthquake hazard at Browns Ferry Nuclear Plant, a study was made of the known seismic history of a large surrounding area. This study was greatly facilitated by research carried on over a period of more than three decades on the seismicity of the southeastern United States in general and the Tennessee Valley region in particular. The more active areas are as follows (TVA, Amendment 25.3, Section 2.5.3.2):

a. Mississippi Valley, especially the New Madrid region of Missouri, Arkansas, Tennessee and Kentucky. The area has been the center of a few great earthquakes and very numerous lighter shocks which are still occurring at intervals. The New Madrid region is about 200 miles northwest of the plant site.
b. The Lower Wabash Valley of Indiana and Illinois. This area has been the center of several moderately strong earthquakes, some of which were felt as far south as Tennessee. The Lower Wabash Valley is about 225 miles north-northwest from the plant site.
c. Charleston area, South Carolina. One of the country's greatest earthquakes was centered in the Charleston area. Many other light-to-moderate earthquakes have occurred in this area and the activity has continued to the present time. Charleston is about 420 miles east of the plant site.
d. The Southern Appalachian area of western North Carolina and eastern Tennessee. Light-to-moderate earthquakes occur in this area at an average frequency of one or two per year. This area is centered about 200 miles east of Decatur.

In addition to these areas, shocks of light-to-moderate intensity have occurred at many other localities in the southeastern states at various distances from Browns Ferry Nuclear Plant. At many of these localities, only a few light-to-moderate shocks from widely scattered centers are known. (TVA, Amendment 25.3, Section 2.5.3.2)

Browns Ferry Nuclear Plant is founded on a thick succession of essentially horizontal sedimentary rocks. The site is 16.5 miles away from the nearest known inactive fault and approximately 200 miles from the New Madrid region of the Mississippi Valley. Since the site area is very low on the southeastern flank of the Nashville structural dome, it has undergone no tectonic movement except simple uplift. This movement probably ceased at the close of the Paleozoic Era. (TVA, Amendment 25.3, Section 2.5.3.3)

The site is underlain by massive formations of bedrock, thus providing adequate foundations for all plant structures. Browns Ferry Nuclear Plant was designed using a conservative assumption that a seismic event at an unstated location could cause a response with an intensity VII on the Modified Mercalli Intensity Scale of 1931 at the plant site. Thus, the design of structures and equipment important to the plant safety features was based on a horizontal ground motion due to a peak acceleration of 0.10g. In addition, the design is such that the plant can be safely shut down during a peak horizontal ground acceleration of 0.20g. (TVA, Amendment 25.3, Section 2.5.4)

2.1 Regional and Local Geology

Browns Ferry Nuclear Plant area lies on the southeastern flank of the Nashville structural dome where it merges into the foreland slope of the Appalachian geosyncline. Throughout most of the Paleozoic Era the region was at or slightly below sea level. During this time more than 5,000 ft of limestone, dolomite, and shale were deposited. Since the end of the Paleozoic Era, some 250,000,000 years ago, the area has been above sea level and has been subjected to numerous cycles of erosion resulting in a general peneplanation. During its history this immediate region has been one of little structural deformation. Major folds and faults are entirely absent. The rock strata are only slightly warped with regional dips of less than 1 degree to the southeast away from the Nashville dome and toward the foreslope of the Appalachian geosyncline. (TVA, Amendment 25.3, Section 2.5.2.3.1)

The low plateau on which Browns Ferry Nuclear Plant lies is underlain by near-horizontal limestone strata of Mississippian age having an aggregate thickness of slightly over 1,000 ft. In ascending order the formations and their maximum thicknesses, according to the Alabama Geological Survey, are: Fort Payne, 207 ft; Tuscumbia, 200 ft; Ste. Genevieve, 43 ft; Bethel, 40 ft; Gasper, 160 ft; Cypress, 7 ft; Golconda, 70 ft; Hartselle, 200 ft; and Bangor, 90 ft. Bedrock is mantled by varying thicknesses of cherty clay, silt, sand, and gravel of residual and alluvial origin. (TVA, Amendment 25.3, Section 2.5.2.3.2)

The only formations involved directly in the site area are the unconsolidated materials overlying bedrock and the Tuscumbia limestone and the Fort Payne Formation. A brief description of each of these follows. (TVA, Amendment 25.3, Section 2.5.2.3.2)

Unconsolidated Deposits – Within the site area bedrock is mantled by an average thickness of 54 ft of red and yellow clay containing some residual chert boulders and lenses of sand and gravel. This material varies in thickness from a known minimum of 41 ft to a known maximum of 69 ft. (TVA, Amendment 25.3, Section 2.5.2.3.2)

Tuscumbia Limestone – Only the lower 50 ft of the Tuscumbia formation was encountered at Browns Ferry Nuclear Plant. The Tuscumbia is characterized by medium-to-thick beds of light-gray, medium-to-coarse-crystalline, fossiliferous limestone. In as much as the Tuscumbia Limestone is a relatively pure limestone, it is more affected by solution (than the Fort Payne Formation). Practically all the cavities encountered at the site were developed in this formation. (TVA, Amendment 25.3, Section 2.5.2.3.2)

Fort Payne Formation - The maximum known thickness of the Fort Payne formation in northern Alabama is slightly over 200 ft. At Browns Ferry Nuclear Plant the total thickness, penetrated in one drill hole, is 145 ft. The formation consists of medium-bedded, medium to dark gray, silty dolomite and siliceous limestone with a few thin horizons of shale. Near the top of the formation, some of the beds are cherty and some of the cores showed zones which were slightly asphaltic. The most distinguishing lithologic feature is the presence of quartz-and calcite-filled vugs up to 1 inch in diameter. The silty, siliceous nature of the Fort Payne formation inhibits the development of solution cavities and very few were found in cores drilled from this formation. In general, excavation grades for the major structures of Browns Ferry Nuclear Plant were set in the Fort Payne formation. (TVA, Amendment 25.3, Section 2.5.2.3.2)

2.2 Probabilistic Seismic Hazard Analysis

2.2.1 Probabilistic Seismic Hazard Analysis Results

In accordance with the 50.54(f) letter (U.S. NRC, 2012a) and following the guidance in the SPID (EPRI, 2013a), a probabilistic seismic hazard analysis (PSHA) was completed using the recently developed Central and Eastern United States Seismic Source Characterization (CEUS-SSC) for Nuclear Facilities (CEUS-SSC, 2012) together with the updated Electric Power Research Institute Ground-Motion Model (GMM) for the Central and Eastern United States (CEUS) (EPRI, 2013b). For the PSHA, a lower-bound moment magnitude of 5.0 was used, as specified in the 50.54(f) letter (U.S. NRC, 2012a). (EPRI, 2014)

For the PSHA, the CEUS-SSC background seismic sources out to a distance of 400 miles (640 km) around Browns Ferry Nuclear Plant were included. This distance exceeds the 200 mile (320 km) recommendation contained in Reg. Guide 1.208 (U.S. NRC, 2007a) and was chosen for completeness. Background sources included in this site analysis were the following (EPRI, 2014):

1. Extended Continental Crust—Atlantic Margin (ECC_AM)
2. Extended Continental Crust—Gulf Coast (ECC_GC)
3. Gulf Highly Extended Crust (GHEX)
4. Illinois Basin Extended Basement (IBEB)
5. Mesozoic and younger extended prior – narrow (MESE-N)
6. Mesozoic and younger extended prior – wide (MESE-W)
7. Midcontinent-Craton alternative A (MIDC_A)
8. Midcontinent-Craton alternative B (MIDC_B)
9. Midcontinent-Craton alternative C (MIDC_C)
10. Midcontinent-Craton alternative D (MIDC_D)
11. Non-Mesozoic and younger extended prior – narrow (NMESE-N)
12. Non-Mesozoic and younger extended prior – wide (NMESE-W)
13. Paleozoic Extended Crust narrow (PEZ_N)
14. Paleozoic Extended Crust wide (PEZ_W)
15. Reelfoot Rift (RR)
16. Reelfoot Rift including the Rough Creek Graben (RR-RCG)
17. Study region (STUDY_R)

For sources of large magnitude earthquakes (designated Repeated Large Magnitude Earthquake (RLME) sources) in NUREG-2115 (CEUS-SSC, 2012) modeled for the CEUS-SSC, the following sources lie within 1,000 km of the site and were included in the analysis (EPRI, 2014):

1. Charleston
2. Commerce
3. Eastern Rift Margin Fault northern segment (ERM-N)
4. Eastern Rift Margin Fault southern segment (ERM-S)
5. Marianna
6. Meers
7. New Madrid Fault System (NMFS)
8. Wabash Valley

For each of the above background and RLME sources, the mid-continent version of the updated CEUS EPRI GMM was used. (EPRI, 2014)

2.2.2 Base Rock Seismic Hazard Curves

Consistent with the SPID (EPRI, 2013a), base rock seismic hazard curves are not provided as the site amplification approach referred to as Method 3 has been used. Seismic hazard curves are shown below in Figure 2.3.7-1. (EPRI, 2014)

2.3 Site Response Evaluation

Following the guidance contained in Seismic Enclosure 1 of the 50.54(f) request for information (U.S. NRC, 2012a) and in the SPID (EPRI, 2013a) for nuclear power plant sites that are not founded on hard rock (defined as 2.83 km/sec), a site response analysis was performed for Browns Ferry Nuclear Plant. (EPRI, 2014)

2.3.1 Description of Subsurface Material

Browns Ferry Nuclear Plant is located on the northern shore of Wheeler Reservoir in Limestone County, in the northern part of Alabama. The site is located near the southern margin of the Highland River section of the Interior Low Plateau physiographic province. The plant is located on an old river terrace surface developed by the Tennessee River. (EPRI, 2014)

The information used to create the site geologic profile at Browns Ferry Nuclear Plant is shown in Tables 2.3.1-1 and 2.3.1-2. This profile was developed using information documented in AMEC (2013). As indicated in Table 2.3.1-1, the SSE Control Point is at a depth of 52 ft (16 m). (EPRI, 2014)

The site is located on limestone of Lower Mississippian age (Fort Payne Chert; Table 2.3.1-2) which is about 200 ft (61 m) thick at the site. It consists of a silty dolomite and siliceous limestone with a few thin shale horizons. The best estimate shear-wave velocity for the Fort Payne Chert is 9,500 ft/s (2,895 m/s) at the depth of the SSE Control Point (Table 2.3.1-2). In Table 2.3.1-2 the shear-wave velocities at greater depths range from 7,000 ft/sec (2,133 m/s) to 9,500 ft/s (2,895 m/s). There is about 3,973 ft (1,211 m) of firm Paleozoic sedimentary rocks which overlie hard basement rock, which is assumed to occur at the top of the Rome Formation beneath the site. (EPRI, 2014)

Table 2.3.1-1. Summary of Site Geotechnical Profile for Browns Ferry Nuclear Plant (AMEC, 2013).(EPRI, 2014)

Depth (ft)	Soil/Rock Description	Density (lb/ft³)	Measured V_s (ft/s)	Recommended V_s for Analyses (ft/s)	G_{max} (lb/ft²)	G/G_{max} vs. Shear Strain	Damping Ratio vs. Shear Strain
0	Ground Surface Elev. 565	–	–	–	–	–	–
0 – 50	Alluvial Clays, Silts over Residual Clays, Silts*	120	700 – 1,800 Average 1,067	1,050	4,000,000	Use Watts Bar FSAR Figure 2.5-233E	Use Watts Bar FSAR Figure 2.5-233F
50 – 52	Dolomite and Limestone	165	---	8,000**	330,000,000	1	No Change
52	**Deepest Structure Foundation Control Point – SSE GMRS**	–	–	–	–	–	–
52 – 100	Dolomite and Limestone	165	–	8,000	330,000,000	1	No Change
100 – 200	Fossiliferous Chert	165	–	8,000	330,000,000	1	No Change

Notes: * Replaced with engineered backfill for safety-related structures
** Calculated from laboratory measured Shear Modulus, G

Table 2.3.1-2. Summary of Geologic Profile Interpolated to Basement for Browns Ferry Nuclear Plant (AMEC, 2013). (EPRI, 2014)

Depth (ft)	**Soil/Rock Description***	**Rock Formation**	**Best Estimate V_s (ft/s)****	**Lower Range V_s (ft/s)*****	**Upper Range V_s (ft/s)*****
0 – 50	Overburden, alluvial clays, silts over residual clays, silts. Thickness 0 to 50 ft.	Mt – Tuscumbia Limestone	1,050****	700****	1,800****
50 – 250	Limestone, light gray, thin- to medium-bedded, siliceous with nodules of light to dark gray fossiliferous chert; lower part of unit locally siliceous dark gray shale. Thickness 100 to 200 ft.	Mfp – Fort Payne Chert	9,500	7,600	9,285
250 – 325	Shale, black to gray, carbonaceous, radioactive, pyritiferous, fissile. Weathers to a greenish gray soil. Thickness 0 to 66 ft.	Dc – Chattanooga Shale	7,000	5,600	8,750
325 – 1,025	Shale and siltstone with thin limestone, gray to reddish-gray, contains one or more hematite rich ore beds in lower half; shale and siltstone interbedded with light green or gray, thick-bedded sandstone in upper half. Thick, light gray limestone unit near middle of interval. Thickness 300 to 700 ft.	Srm – Red Mountain Formation	7,000	5,600	8,750
1,025 – 1,225	Limestone, light to dark gray, thin- to medium-bedded, fine grained, highly argillaceous and fossiliferous, interbedded with variegated greenish-gray and maroon calcareous shale. Thickness about 200 ft.	Os – Sequatchie Formation	9,500	6,050	9,285

Table 2.3.1-2. Summary of Geologic Profile Interpolated to Basement for Browns Ferry Nuclear Plant (AMEC, 2013), Continued. (EPRI, 2014)

Depth (ft)	**Soil/Rock Description***	**Rock Formation**	**Best Estimate V_s (ft/s)****	**Lower Range V_s (ft/s)*****	**Upper Range V_s (ft/s)*****
1,225 – 1,825	Limestone, light to medium gray, cryptocrystalline to coarsely crystalline, slabby to medium-bedded, argillaceous in part; numerous thin bentonite layers. Bentonites separate Unit I from Unit II. Thickness 200 to 600 ft.	Oc – Chickamauga Group	9,500	6,050	9,285
1,825 – 3,425	Dolomite and minor limestone, very siliceous, light- to dark-gray, fine- to coarse-grained, thin- to thick- bedded, weathers to cherty rubble. Thickness about 2,600 ft.	OЄk – Knox Group, Undifferentiated	7,000	4,460	9,285
3,425 – 4,025	Shale, gray and greenish-gray, thin-bedded; siltstone, gray, thin- bedded, glauconitic and calcareous; limestone, thin- bedded, edgewise conglomerates consisting of dolomitic rip-up clasts throughout the middle and upper part of the formation. Lower part consists of interbedded siltstone, and shale, gray and greenish- gray, thin-bedded, glauconitic, micaceous, commonly bioturbated, a few marine shell fossils found.	Єc – Conasauga Group Lower Undivided	7,000	4,460	9,285

Table 2.3.1-2. Summary of Geologic Profile Interpolated to Basement for Browns Ferry Nuclear Plant (AMEC, 2013), Continued. (EPRI, 2014)

Depth (ft)	Soil/Rock Description*	Rock Formation	Best Estimate V_s (ft/s)**	Lower Range V_s (ft/s)***	Upper Range V_s (ft/s)***
4,025 – 4,200	Sandstone, reddish-brown, greenish-gray, light-brown, olive, fine-to medium-grained, thin-to thick-bedded, glauconitic, micaceous; interbedded with shale and siltstone, reddish- brown, olive greenish-gray, light-brown, thin-bedded, micaceous, bioturbated; dolomite and dolomoitic limestone may also be present; thrust fault at base, estimated exposed thickness shown.	Ꞓr – Rome Formation	10,000	6,370	9,285
> 4,200	-	Basement	12,000	7,640	9,285

*Note: Rock Descriptions obtained from GSA (1969) and Lemiszki et al. (2008).

**Note: These values were based on Spectral-Analysis-of-Surface-Waves (SASW) testing by Dr. Ken Stokoe at the Watts Bar Nuclear Plant site, which consists of similar rock formation to base these values upon. Ivan Wong from URS assisted Dr. Stokoe and AMEC in developing a lognormal average for the best estimate.

***Note: The lower and upper ranges were based on the best estimate, with the upper range constrained not to exceed 9,285 ft/s. For depths of 0 – 50 ft, these values were calculated using a V_s value for limestone of 9,500 ft/s and a certainty of 1.25. For depths of 50 – 1,000 ft, these values were calculated using a certainty of 1.25. For depths of 1,000 ft to basement, these values were calculated using a certainty of 1.57.

****Note: These values were not determined by the same methods outlined in **Note and ***Note. These values were obtained from the previous geotechnical exploration shown in Table 1(AMEC, 2013).

The following description of the Paleozoic sequence is extracted from AMEC (2013): (EPRI, 2014)

> "The low plateau on which the Browns Ferry site lies is underlain by near-horizontal limestone strata of Mississippian age having an aggregate thickness of slightly over 1,000 ft. The regional structure in the Browns Ferry area is controlled by the Nashville dome. The area lies on the southeast flank of this dome and the regional dip is a degree or less to the southeast."

"In the immediate site area, the beds of the Tuscumbia Limestone and Fort Payne Chert formations are essentially horizontal. As is to be expected in near-horizontal strata, bedrock is cut by a pattern of near-vertical joints. Close to the surface of bedrock, solution channels have developed along these joints especially in the Tuscumbia Limestone. At depth, however, in the less soluble Fort Payne Chert, the joints are tight and most are cemented with calcite. The Browns Ferry Nuclear Plant is underlain by two geologic formations which are outlined by the FSAR."

"Tuscumbia Limestone – Only the lower 50 ft of the Tuscumbia Limestone formation was encountered at the Browns Ferry site. The Tuscumbia Limestone is characterized by medium-to-thick beds of light-gray, medium-to-coarse crystalline, fossiliferous limestone. In as much as the Tuscumbia Limestone is a relatively pure limestone, it is more affected by solution (than the Fort Payne Chert Formation). Practically all the cavities encountered at the site were developed in this formation."

"Fort Payne Formation – The maximum known thickness of the Fort Payne formation in northern Alabama is slightly more than 200 ft. At the Browns Ferry site, the total thickness, penetrated in one drill hole, is 145 ft. The formation consists of medium-bedded, medium to dark gray, silty dolomite and siliceous limestone with a few thin horizons of shale. Near the top of the formation, some of the beds are cherty and some of the cores showed zones which were slightly asphaltic. The most distinguishing lithologic feature is the presence of quartz-and calcite-filled vugs up to 1 inch in diameter. The silty, siliceous nature of the Fort Payne formation inhibits the development of solution cavities and very few were found in cores drilled from this formation."

2.3.2 Development of Base Case Profiles and Nonlinear Material Properties

Table 2.3.2-2 (AMEC, 2013) shows the recommended shear-wave velocities and unit weights along with depths and corresponding stratigraphy from the surface to basement. AMEC (2013) states that the SSE control point is at depth of 52 ft (16 m) near the top of the Fort Payne Chert with an assumed shear-wave velocity adopted from Watts Bar Nuclear Plant of 9,500 ft/s (2,895 m/s). Deeper shear-wave velocity values were also taken from Watts Bar Nuclear Plant since it is founded on similar geology. Based on the SPID (EPRI, 2013a), the maximum shear-wave velocity used in the profile development is 9,285 ft/s (2,830 m/s). The depth to reference hard rock (basement) is 3,973 ft (1,211 m). (EPRI, 2014)

Based on the shear-wave velocities that were not measured at the site but obtained from a site with similar geology, a scale factor of 1.57 was adopted to reflect upper and lower range base-cases. The scale factor of 1.57 reflects a $\sigma_{\mu\ln}$ of about 0.35 based on the SPID (EPRI, 2013a) 10^{th} and 90^{th} fractiles which implies a 1.28 scale factor on σ_{μ}. (EPRI, 2014)

Using the shear-wave velocities specified in Table 2.3.2-1, three base-case profiles were developed using the scale factor of 1.57. The specified shear-wave velocities were taken as the mean or best estimate base-case profile (P1) with lower- and upper- range base-case profiles P2 and P3. Profiles P1 and P2 extended to hard reference rock at a depth below the SSE control point at 3,973 ft (1,211 m), randomized ±1,192 ft (363 m). For the stiffest profile (P3), upper-range shear-wave velocities exceeded the hard rock value of 9,285 ft/s (2,830 m/s), resulting in adopting P3 as reflecting reference site conditions. The depth randomization reflects ±30% of the depth and was included to provide a realistic broadening of the fundamental resonance at deep sites rather than reflect actual random variations to basement shear-wave velocities across a footprint. (EPRI, 2014)

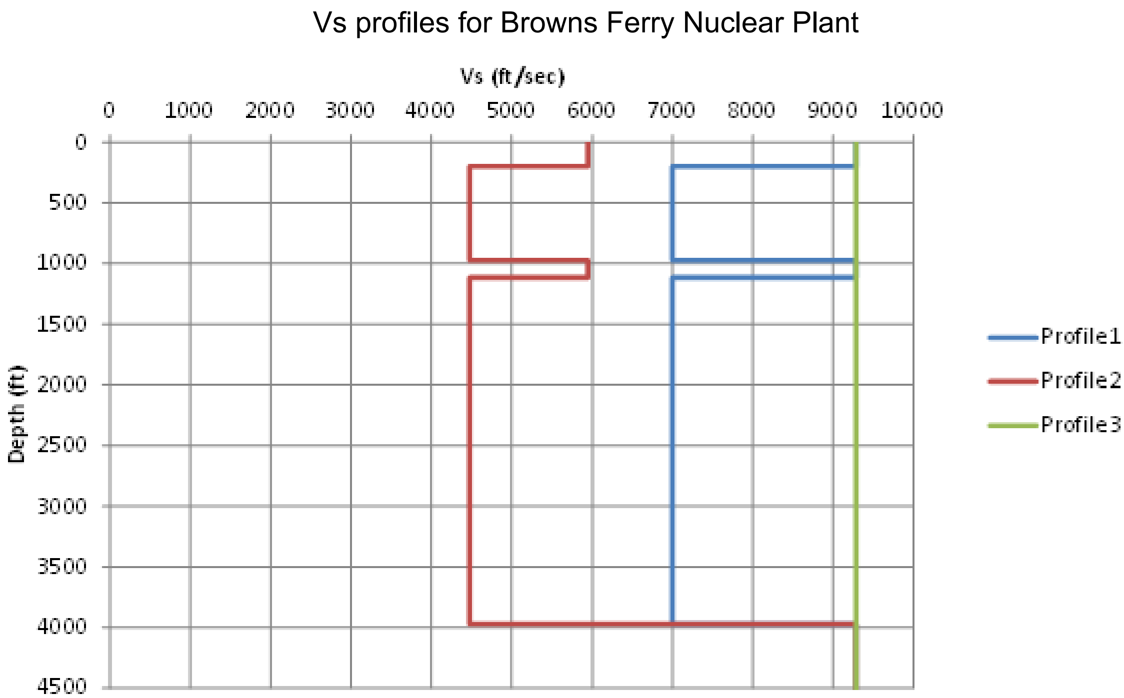


Figure 2.3.2-1. Shear-wave velocity profiles for Browns Ferry Nuclear Plant. (EPRI, 2014)

Table 2.3.2-1. Layer thicknesses, depths, and shear-wave velocities (V_s) for 3 profiles, Browns Ferry Nuclear Plant. (EPRI, 2014)

Profile 1			Profile 2			Profile 3		
thickness (ft)	depth (ft)	V_s (ft/s)	thickness (ft)	depth (ft)	V_s (ft/s)	thickness (ft)	depth (ft)	V_s (ft/s)
	0	9285		0	5942		0	9285
10.0	10.0	9285	10.0	10.0	5942	10.0	10.0	9285
10.0	20.0	9285	10.0	20.0	5942	10.0	20.0	9285
8.0	28.0	9285	8.0	28.0	5942	8.0	28.0	9285
10.0	38.0	9285	10.0	38.0	5942	10.0	38.0	9285
10.0	48.0	9285	10.0	48.0	5942	10.0	48.0	9285
2.0	50.0	9285	2.0	50.0	5942	2.0	50.0	9285
10.0	60.0	9285	10.0	60.0	5942	10.0	60.0	9285
10.0	70.0	9285	10.0	70.0	5942	10.0	70.0	9285
10.0	80.0	9285	10.0	80.0	5942	10.0	80.0	9285
10.0	90.0	9285	10.0	90.0	5942	10.0	90.0	9285
10.0	100.0	9285	10.0	100.0	5942	10.0	100.0	9285
10.0	110.0	9285	10.0	110.0	5942	10.0	110.0	9285
10.0	120.0	9285	10.0	120.0	5942	10.0	120.0	9285
10.0	130.0	9285	10.0	130.0	5942	10.0	130.0	9285
10.0	140.0	9285	10.0	140.0	5942	10.0	140.0	9285
8.0	148.0	9285	8.0	148.0	5942	8.0	148.0	9285
2.0	150.0	9285	2.0	150.0	5942	2.0	150.0	9285
10.0	160.0	9285	10.0	160.0	5942	10.0	160.0	9285
10.0	170.0	9285	10.0	170.0	5942	10.0	170.0	9285
10.0	180.0	9285	10.0	180.0	5942	10.0	180.0	9285
10.0	190.0	9285	10.0	190.0	5942	10.0	190.0	9285
8.0	198.0	9285	8.0	198.0	5942	10.0	200.0	9285
10.0	208.0	7000	10.0	208.0	4480	10.0	210.0	9285
10.0	218.0	7000	10.0	218.0	4480	10.0	220.0	9285
10.0	228.0	7000	10.0	228.0	4480	10.0	230.0	9285
10.0	238.0	7000	10.0	238.0	4480	10.0	240.0	9285
10.0	248.0	7000	10.0	248.0	4480	10.0	250.0	9285
10.0	258.0	7000	10.0	258.0	4480	10.0	260.0	9285
10.0	268.0	7000	10.0	268.0	4480	10.0	270.0	9285
10.0	278.0	7000	10.0	278.0	4480	10.0	280.0	9285
10.0	288.0	7000	10.0	288.0	4480	10.0	290.0	9285
10.0	298.0	7000	10.0	298.0	4480	10.0	300.0	9285
10.0	308.0	7000	10.0	308.0	4480	10.0	310.0	9285
10.0	318.0	7000	10.0	318.0	4480	10.0	320.0	9285
10.0	328.0	7000	10.0	328.0	4480	10.0	330.0	9285
10.0	338.0	7000	10.0	338.0	4480	10.0	340.0	9285
10.0	348.0	7000	10.0	348.0	4480	10.0	350.0	9285

Table 2.3.2-1. Layer thicknesses, depths, and shear-wave velocities (V_s) for 3 profiles, Browns Ferry Nuclear Plant, Continued. (EPRI, 2014)

Profile 1			Profile 2			Profile 3		
thickness (ft)	depth (ft)	V_s (ft/s)	thickness (ft)	depth (ft)	V_s (ft/s)	thickness (ft)	depth (ft)	V_s (ft/s)
10.0	358.0	7000	10.0	358.0	4480	10.0	360.0	9285
10.0	368.0	7000	10.0	368.0	4480	10.0	370.0	9285
10.0	378.0	7000	10.0	378.0	4480	10.0	380.0	9285
10.0	388.0	7000	10.0	388.0	4480	10.0	390.0	9285
10.0	398.0	7000	10.0	398.0	4480	10.0	400.0	9285
10.0	408.0	7000	10.0	408.0	4480	10.0	410.0	9285
10.0	418.0	7000	10.0	418.0	4480	10.0	420.0	9285
10.0	428.0	7000	10.0	428.0	4480	10.0	430.0	9285
10.0	438.0	7000	10.0	438.0	4480	10.0	440.0	9285
10.0	448.0	7000	10.0	448.0	4480	10.0	450.0	9285
10.0	458.0	7000	10.0	458.0	4480	10.0	460.0	9285
10.0	468.0	7000	10.0	468.0	4480	10.0	470.0	9285
10.0	478.0	7000	10.0	478.0	4480	10.0	480.0	9285
10.0	488.0	7000	10.0	488.0	4480	10.0	490.0	9285
10.0	498.0	7000	10.0	498.0	4480	10.0	500.0	9285
40.8	538.8	7000	40.8	538.8	4480	40.8	540.8	9285
40.8	579.6	7000	40.8	579.6	4480	40.8	581.6	9285
40.8	620.4	7000	40.8	620.4	4480	40.8	622.4	9285
40.8	661.2	7000	40.8	661.2	4480	40.8	663.2	9285
40.8	702.0	7000	40.8	702.0	4480	40.8	704.0	9285
40.8	742.8	7000	40.8	742.8	4480	40.8	744.8	9285
40.8	783.6	7000	40.8	783.6	4480	40.8	785.6	9285
40.8	824.4	7000	40.8	824.4	4480	40.8	826.4	9285
40.8	865.2	7000	40.8	865.2	4480	40.8	867.2	9285
40.8	906.0	7000	40.8	906.0	4480	40.8	908.0	9285
67.0	973.0	7000	67.0	973.0	4480	409.2	1317.1	9285
71.9	1044.9	9285	71.9	1044.9	5942	409.2	1726.3	9285
71.9	1116.7	9285	71.9	1116.7	5942	409.2	2135.5	9285
401.4	1518.1	7000	401.4	1518.1	4480	409.2	2544.7	9285
408.9	1927.0	7000	408.9	1927.0	4480	409.2	2953.9	9285
409.2	2336.2	7000	409.2	2336.2	4480	409.2	3363.0	9285
409.2	2745.3	7000	409.2	2745.3	4480	409.2	3772.2	9285
409.2	3154.5	7000	409.2	3154.5	4480	409.2	4181.4	9285
409.2	3563.7	7000	409.2	3563.7	4480	409.2	4590.6	9285
409.2	3972.9	7000	409.2	3972.9	4480	409.2	4999.8	9285
3280.8	7253.7	9285	3280.8	7253.7	9285	3280.8	8280.6	9285

2.3.2.1 Shear Modulus and Damping Curves

No site-specific nonlinear dynamic material properties were determined for the firm rock materials in the initial siting of Browns Ferry Nuclear Plant. The rock material over the upper 500 ft (152 m) was assumed to have behavior that could be modeled as either linear or non-linear. To represent this potential for either case in the upper 500 ft of firm rock at Browns Ferry Nuclear Plant, two sets of shear modulus reduction and hysteretic damping curves were used. Consistent with the SPID (EPRI, 2013a), the EPRI rock curves (model M1) were considered to be appropriate to represent the upper range nonlinearity likely in the materials at this site and linear analyses (model M2) was assumed to represent an equally plausible alternative rock response across loading level. For the linear analyses, the low strain damping from the EPRI rock curves were used as the constant damping values in the upper 500 ft. (EPRI, 2014)

2.3.2.2 Kappa

Base-case kappa estimates were determined using Section B-5.1.3.1 of the SPID (EPRI, 2013a) for a firm CEUS rock site. Kappa for a firm rock site with at least 3,000 ft (1 km) of sedimentary rock may be estimated from the average S-wave velocity over the upper 100 ft (V_{s100}) of the subsurface profile while for a site with less than 3,000 ft (1 km) of firm rock, kappa may be estimated with a Q_s of 40 below 500 ft combined with the low strain damping from the EPRI rock and or soil curves and an additional kappa of 0.006 s for the underlying hard rock. For Browns Ferry Nuclear Plant, with about 3,973 ft (1,211 m) of firm rock below the SSE, kappa estimates were based on the average shear-wave velocity over the top 100 ft (30 m) of the three base-case profiles P1, P2, and P3. For the three profiles the corresponding shear-wave velocities were: 9,285 ft/s (2,830 m/s), 5,914 ft/s (1,802 m/s), and 9,285 ft/s (2,830 m/s) with corresponding kappa estimates of 0.006 s, 0.012 s, and 0.006 s. The range in kappa about the average base-case value of 0.008 s is roughly 1.4 and was considered to adequately reflect epistemic uncertainty in low strain damping (kappa) for the profile. Additionally, for very stiff firm rock profiles, contributions to epistemic uncertainty in low strain kappa are assumed to be incorporated in the reference rock hazard. Values for kappa as well as the weights used for the site response analyses are presented below in Table 2.3.2-2. (EPRI, 2014)

Table 2.3.2-2. Kappa Values and Weights Used for Site Response Analyses. (EPRI, 2014)

Velocity Profile	Kappa(s)
P1	0.006
P2	0.012
P3	0.006
	Weights
P1	0.4
P2	0.3
P3	0.3
G/G_{max} and Hysteretic Damping Curves	
M1	0.5
M2	0.5

2.3.3 Randomization of Base Case Profiles

To account for the aleatory variability in dynamic material properties that is expected to occur across a site at the scale of a typical nuclear facility, variability in the assumed shear-wave velocity profiles has been incorporated in the site response calculations. For Browns Ferry Nuclear Plant,random shear-wave velocity profiles were developed from the base case profiles shown in Figure 2.3.2-1. Consistent with the discussion in Appendix B of the SPID (EPRI, 2013a), the velocity randomization procedure made use of random field models which describe the statistical correlation between layering and shear wave velocity. The default randomization parameters developed in Toro (1997) for United States Geological Survey (USGS) “A” site conditions were used for this site. Thirty random velocity profiles were generated for each base case profile. These random velocity profiles were generated using a natural log standard deviation of 0.25 over the upper 50 ft and 0.15 below that depth. As specified in the SPID (EPRI, 2013a), correlation of shear-wave velocity between layers was modeled using the footprint correlation model. In the correlation model, a limit of ±2 standard deviations about the median value in each layer was assumed for the limits on random velocity fluctuations. (EPRI, 2014)

2.3.4 Input Spectra

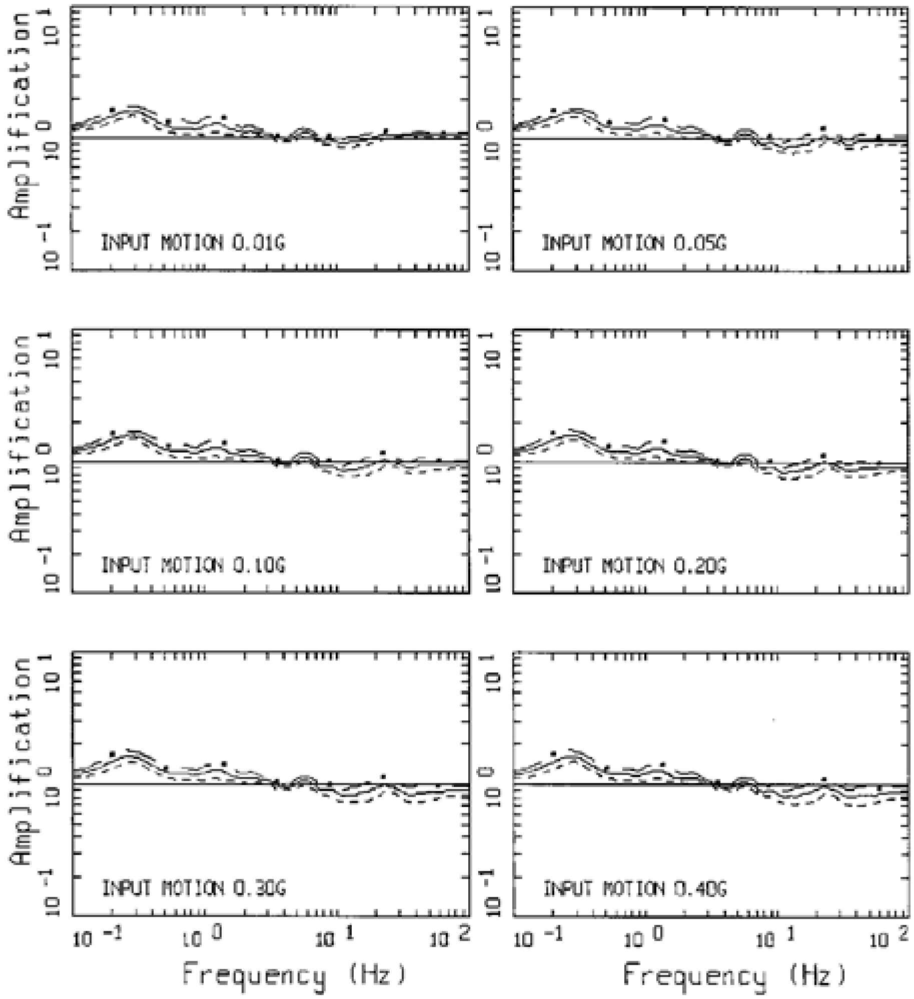
Consistent with the guidance in Appendix B of the SPID (EPRI, 2013a), input Fourier amplitude spectra were defined for a single representative earthquake magnitude (**M** 6.5) using two different assumptions regarding the shape of the seismic source spectrum (single-corner and double-corner). A range of 11 different input amplitudes (median Peak Ground Accelerations (PGAs) ranging from 0.01 to 1.5g) were used in the site response analyses. The characteristics of the seismic source and upper crustal attenuation properties assumed for the analysis of Browns Ferry Nuclear Plant were the same as those identified in Tables B-4, B-5, B-6 and B-7 of the SPID (EPRI, 2013a) as appropriate for typical CEUS sites. (EPRI, 2014)

2.3.5 Methodology

To perform the site response analyses for Browns Ferry Nuclear Plant, a random vibration theory (RVT) approach was employed. This process utilizes a simple, efficient approach for computing site-specific amplification functions and is consistent with existing NRC guidance and the SPID (EPRI, 2013a). The guidance contained in Appendix B of the SPID (EPRI, 2013a) on incorporating epistemic uncertainty in shear-wave velocities, kappa, non-linear dynamic properties and source spectra for plants with limited at-site information was followed for Browns Ferry Nuclear Plant. (EPRI, 2014)

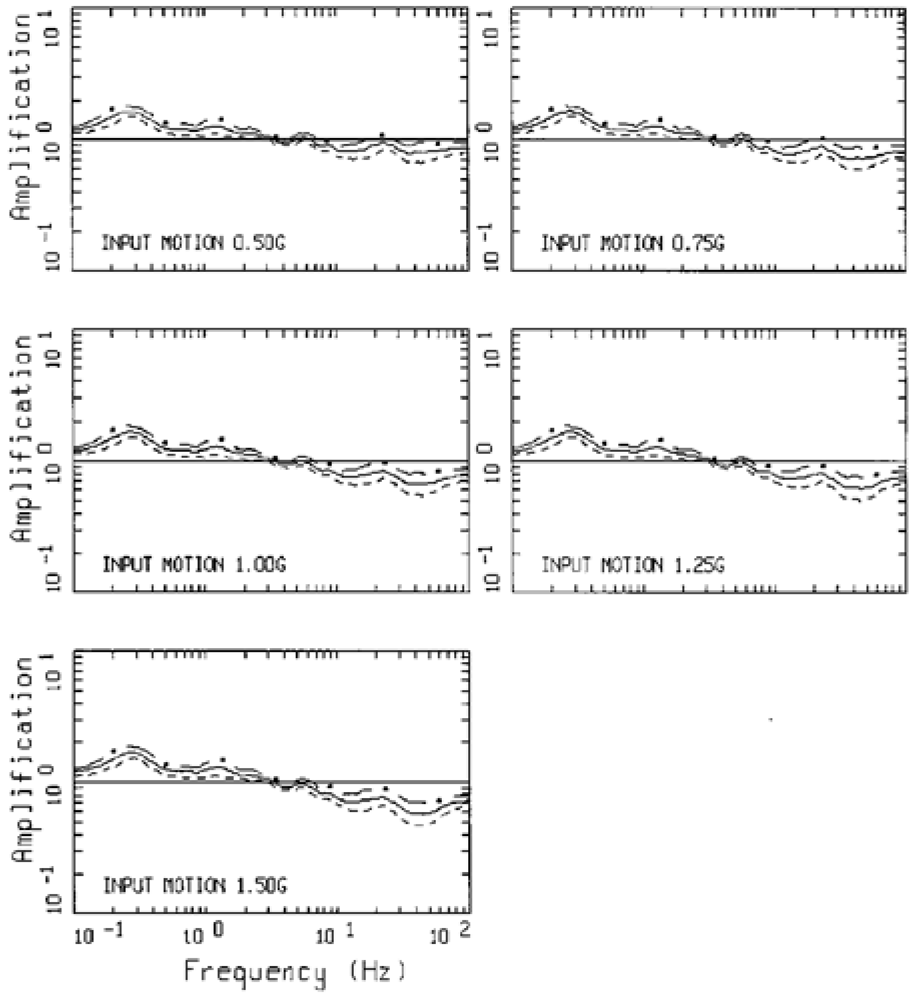
2.3.6 Amplification Functions

The results of the site response analysis consist of amplification factors (5%-damped pseudo-absolute response spectra) which describe the amplification (or de-amplification) of hard reference rock motion as a function of frequency and input reference rock amplitude. The amplification factors are represented in terms of a median amplification value and an associated standard deviation (sigma) for each oscillator frequency and input rock amplitude. Consistent with the SPID (EPRI, 2013a) a minimum median amplification value of 0.5 was employed in the present analysis. Figure 2.3.6-1 illustrates the median and ±1 standard deviation in the predicted amplification factors developed for the eleven loading levels parameterized by the median reference (hard rock) peak acceleration (0.01g to 1.50g) for profile P1 and EPRI (EPRI, 2013a) rock G/G_{max} and hysteretic damping curves. The variability in the amplification factors results from variability in shear-wave velocity, depth to hard rock, and modulus reduction and hysteretic damping curves. To illustrate the effects of nonlinearity at Browns Ferry Nuclear Plant firm rock site, Figure 2.3.6-2 shows the corresponding amplification factors developed with linear site response analyses (model M2). Between the linear and nonlinear (equivalent-linear) analyses, Figures 2.3.6-1 and Figure 2.3.6-2 respectively show only a minor difference for all frequencies at the 0.4g loading level and below. Above about the 0.4g loading level, the differences increase but only above about 5 to 10 Hz. Tabular data for Figure 2.3.6-1 and Figure 2.3.6-2 is provided for information only in Appendix A. (EPRI, 2014)



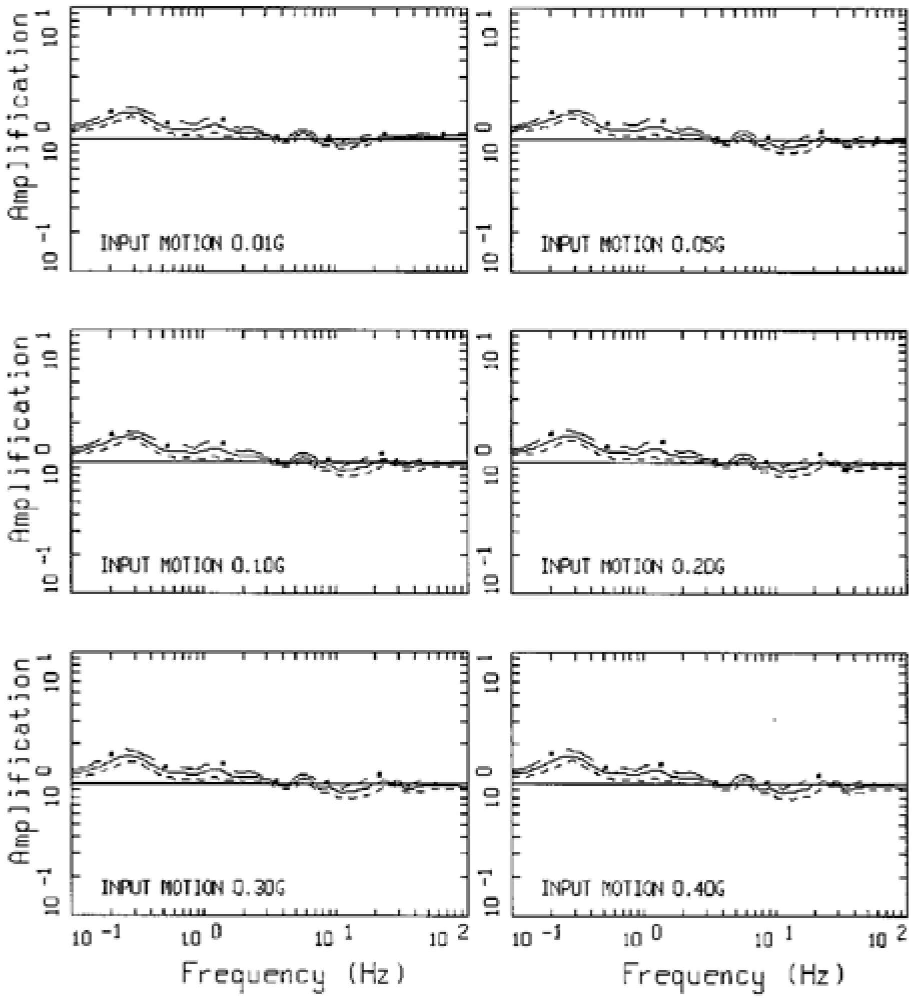
AMPLIFICATION, BROWNS FERRY, M1P1K1
M 6.5, 1 CORNER: PAGE 1 OF 2

Figure 2.3.6-1. Example suite of amplification factors (5%-damping pseudo-absolute acceleration spectra) developed for the mean base-case profile (P1), EPRI rock modulus reduction and hysteretic damping curves (model M1), and base-case kappa (K1) at eleven loading levels of hard rock median peak acceleration values from 0.01g to 1.50g. **M** 6.5 and single-corner source model (EPRI, 2013a). (EPRI, 2014)



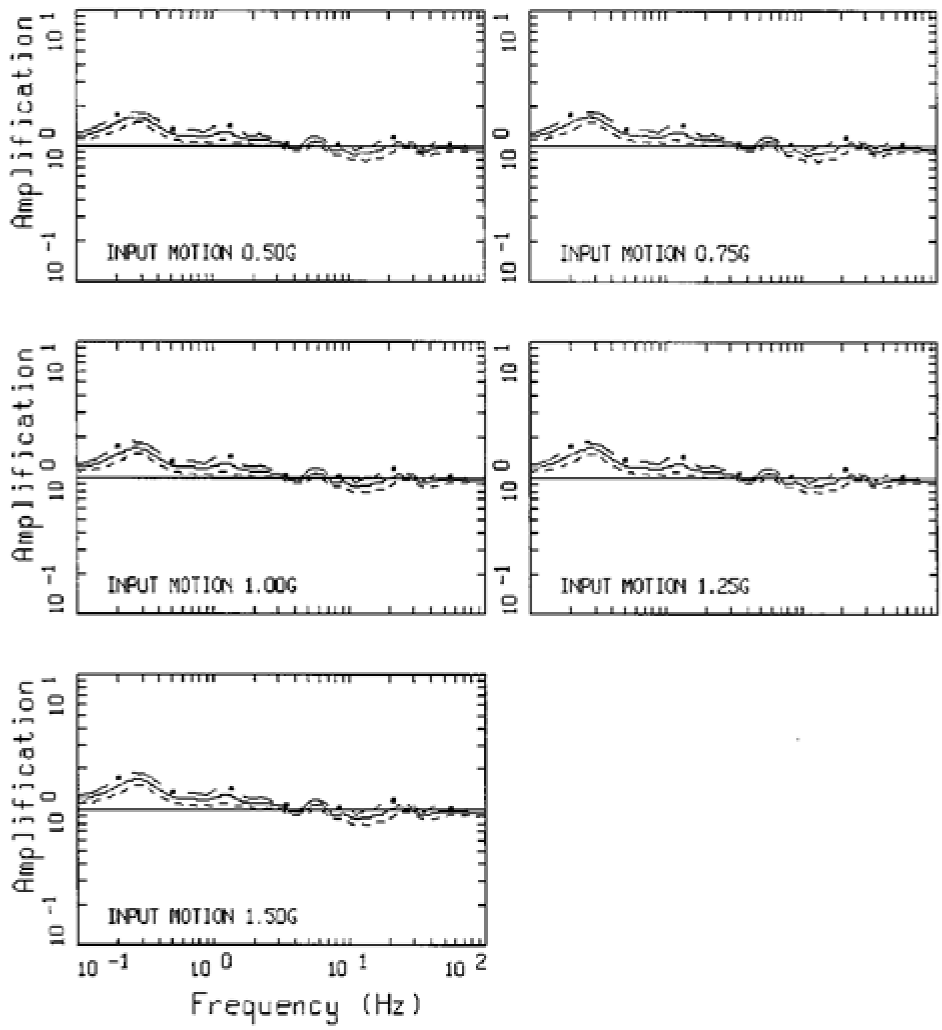
AMPLIFICATION, BROWNS FERRY, M1P1K1
M 6.5, 1 CORNER: PAGE 2 OF 2

Figure 2.3.6-1.(cont.)



AMPLIFICATION, BROWNS FERRY, M2P1K1
M 6.5, 1 CORNER: PAGE 1 OF 2

Figure 2.3.6-2. Example suite of amplification factors (5%-damping pseudo-absolute acceleration spectra) developed for the mean base-case profile (P1), linear site response (model M2), and base-case kappa (K1) at eleven loading levels of hard rock median peak acceleration values from 0.01g to 1.50g. **M** 6.5 and single-corner source model (EPRI, 2013a). (EPRI, 2014)



AMPLIFICATION, BROWNS FERRY, M2P1K1
M 6.5, 1 CORNER: PAGE 2 OF 2

Figure 2.3.6-2.(cont.)

2.3.7 Control Point Seismic Hazard Curves

The procedure to develop probabilistic site-specific control point hazard curves used in the present analysis follows the methodology described in Section B-6.0 of the SPID (EPRI, 2013a). This procedure (referred to as Method 3) computes a site-specific control point hazard curve for a broad range of spectral accelerations given the site-specific bedrock hazard curve and site-specific estimates of soil or soft-rock response and associated uncertainties. This process is repeated for each of the seven spectral frequencies for which ground motion equations are available. The dynamic response of the materials below the control point was represented by the frequency- and amplitude-dependent amplification functions (median values and standard deviations) developed and described in the previous section. The resulting control point mean hazard curves for Browns Ferry Nuclear Plant are shown in Figure 2.3.7-1 for the seven spectral frequencies for which ground motion equations are defined. Tabulated values of mean and fractile seismic hazard curves and site response amplification functions are provided in Appendix A. (EPRI, 2014)

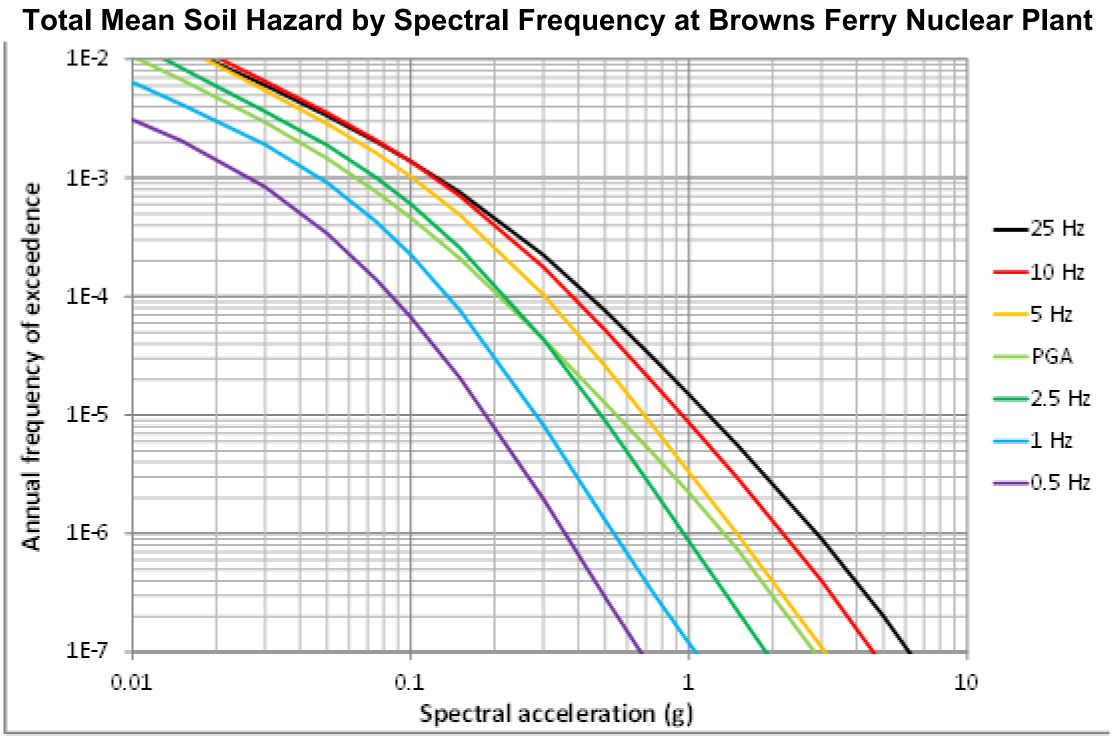


Figure 2.3.7-1. Control point mean hazard curves for spectral frequencies of 0.5, 1.0, 2.5, 5.0, 10, 25 and PGA (100 Hz) at Browns Ferry Nuclear Plant. (EPRI, 2014)

2.4 Control Point Response Spectrum

The control point hazard curves described above have been used to develop Uniform Hazard Response Spectra (UHRS) and the GMRS. The UHRS were obtained through linear interpolation in log-log space to estimate the spectral acceleration at each spectral frequency for the 10^{-4} and 10^{-5} per year hazard levels.

The 10^{-4} and 10^{-5} UHRS, along with a design factor are used to compute the GMRS at the control point using the criteria in Regulatory Guide 1.208 (U.S. NRC, 2007a). Table 2.4-1 shows the UHRS and GMRS accelerations for a range of frequencies. (EPRI, 2014)

Table 2.4-1. UHRS and GMRS for Browns Ferry Nuclear Plant. (EPRI, 2014)

Freq. (Hz)	10^{-4} UHRS (g)	10^{-5} UHRS (g)	GMRS (g)
100	2.08E-01	5.49E-01	2.71E-01
90	2.10E-01	5.55E-01	2.74E-01
80	2.13E-01	5.66E-01	2.79E-01
70	2.22E-01	5.94E-01	2.93E-01
60	2.46E-01	6.67E-01	3.28E-01
50	2.99E-01	8.25E-01	4.04E-01
40	3.63E-01	1.00E+00	4.90E-01
35	3.88E-01	1.06E+00	5.20E-01
30	4.19E-01	1.13E+00	5.57E-01
25	4.39E-01	1.18E+00	5.80E-01
20	4.37E-01	1.15E+00	5.68E-01
15	4.15E-01	1.06E+00	5.27E-01
12.5	3.97E-01	1.00E+00	4.99E-01
10	3.81E-01	9.45E-01	4.73E-01
9	3.71E-01	9.08E-01	4.56E-01
8	3.50E-01	8.45E-01	4.25E-01
7	3.40E-01	8.02E-01	4.05E-01
6	3.38E-01	7.81E-01	3.96E-01
5	3.04E-01	6.91E-01	3.52E-01
4	2.79E-01	6.25E-01	3.19E-01
3.5	2.64E-01	5.91E-01	3.02E-01
3	2.45E-01	5.49E-01	2.80E-01
2.5	2.17E-01	4.83E-01	2.47E-01
2	2.06E-01	4.51E-01	2.31E-01
1.5	1.79E-01	3.85E-01	1.98E-01
1.25	1.62E-01	3.43E-01	1.77E-01
1	1.36E-01	2.83E-01	1.46E-01
0.9	1.26E-01	2.64E-01	1.37E-01
0.8	1.20E-01	2.52E-01	1.30E-01
0.7	1.10E-01	2.34E-01	1.21E-01
0.6	9.78E-02	2.10E-01	1.08E-01
0.5	8.55E-02	1.86E-01	9.54E-02
0.4	6.84E-02	1.49E-01	7.63E-02
0.35	5.98E-02	1.30E-01	6.68E-02
0.3	5.13E-02	1.11E-01	5.72E-02
0.25	4.27E-02	9.29E-02	4.77E-02
0.2	3.42E-02	7.43E-02	3.82E-02
0.15	2.56E-02	5.57E-02	2.86E-02
0.125	2.14E-02	4.64E-02	2.39E-02
0.1	1.71E-02	3.71E-02	1.91E-02

Figures 2.4.-1 shows the control point UHRS and GMRS. (EPRI, 2014)

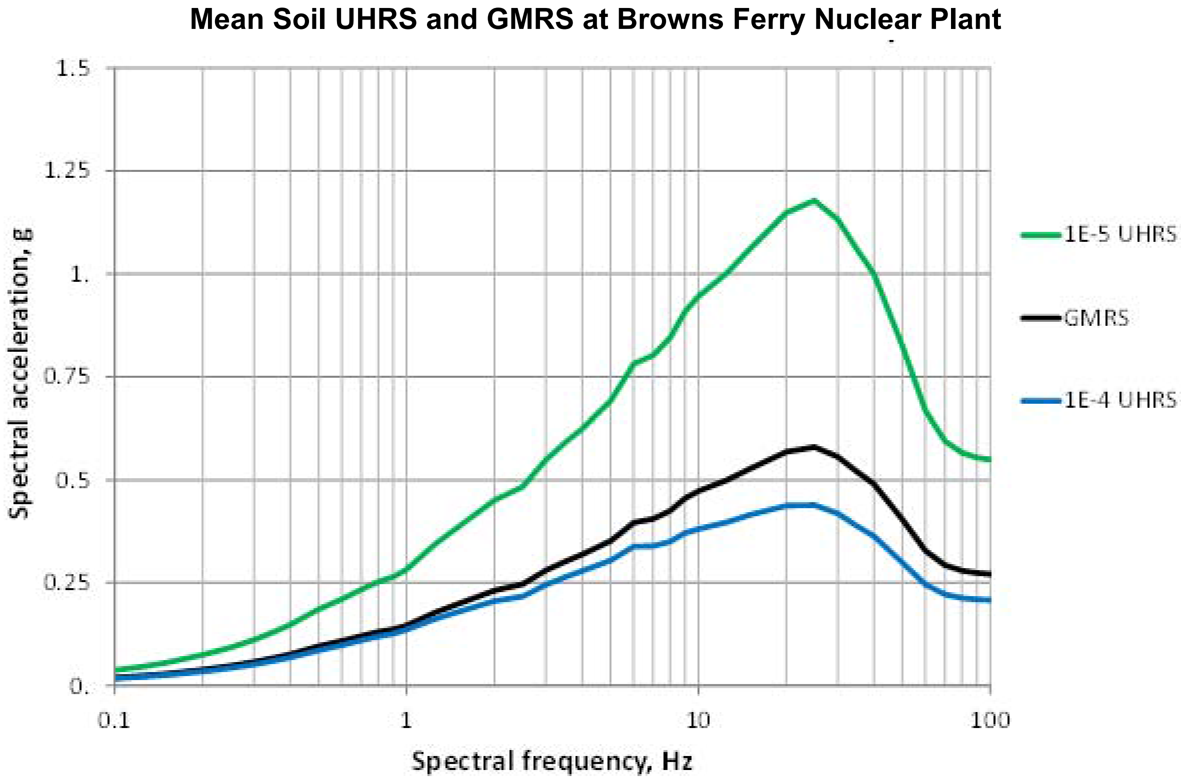


Figure 2.4-1. UHRS for 10^{-4} and 10^{-5} and GMRS at control point for Browns Ferry Nuclear Plant (5%-damped response spectra). (EPRI, 2014).

3.0 Plant Design Basis and Beyond Design Basis Evaluation Ground Motion

The design basis for Browns Ferry Nuclear Plant is identified in the Updated Final Safety Analysis Report. (TVA, Amendment 25.3)

An evaluation for Beyond Design Basis (BDB) ground motions was performed in the Individual Plant Examination of External Events (IPEEE). The IPEEE plant level HCLPF response spectrum is included below for screening purposes.

3.1 Safe Shutdown Earthquake Description of Spectral Shape

The SSE was developed consistent with 10 CFR Part 100, Appendix A through an evaluation of the maximum earthquake potential for the region surrounding the site. Considering the historic seismicity of the site region, Browns Ferry Nuclear Plant was designed using a conservative assumption that a seismic event at an unstated location could cause a response with an intensity VII on the Modified Mercalli Intensity Scale of 1931 at the plant site.

The SSE is defined in terms of a PGA and a design response spectrum. Considering a site intensity of VII, a PGA of 0.20g was estimated. Table 3.1-1 shows the Spectral Acceleration (SA) values as a function of frequency for the 5%-damped horizontal SSE. (EPRI, 2014)

Table 3.1-1. SSE for Browns Ferry Nuclear Plant (AMEC, 2013). (EPRI, 2014)

Freq. (Hz)	100	25	10	5.0	2.5	1.0	0.5
SA (g)	0.20	0.20	0.22	0.30	0.28	0.16	0.12

3.2 Control Point Elevation

The SSE control point elevation is defined at a depth of 52 ft as indicated in Table 2.3.1-1 (EPRI, 2014)

3.3 IPEEE Description and Capacity Response Spectrum

A focused-scope Seismic Margins Assessment (SMA) was performed to support the IPEEE for Browns Ferry Nuclear Plant Units 1, 2 and 3. The results of the IPEEE were submitted to the NRC (TVA, 1996) (TVA, 2005). Results of the NRC review are documented in references (U.S. NRC, 2000) and (U.S. NRC, 2007b).

Browns Ferry Nuclear Plant Units 1, 2 and 3 Seismic IPEEE was performed using the SMA option per the methodology of EPRI NP-6041-SL (EPRI, 1991). With this method, a Seismic Margins Earthquake (SME) was postulated and the items needed for safe shutdown were then evaluated for the SME demand in two success paths (EPRI SMA method). Components and structures that were determined to have sufficient capacity to survive the SME without loss of function were screened out. Items that did not screen were subjected to a more detailed evaluation, including calculation of a High-Confidence-Low-Probability of Failure (HCLPF) PGA for that item. A 0.30g Review Level Earthquake (RLE) level and the NUREG/CR-0098 (U.S. NRC, 1978) median response spectra shape were used. The IPEEE adequacy determination according to SPID Section 3.3.1 (EPRI, 2013a) is included as Appendix B. For IPEEE screening purposes, at Browns Ferry Nuclear Plant Units 1, 2 and 3, the NUREG/CR-0098 (U.S. NRC, 1978) the IPEEE HCLPF Spectrum (IHS) anchored at the RLE of 0.26g will be utilized as discussed in Appendix B.

The 5% damped horizontal IHS spectral accelerations are provided in Table 3.3-1. The SSE and IHS are shown in Figure 3.3-1.

Table 3.3-1. IHS for Browns Ferry Nuclear Plant – Units 1, 2, and 3

Frequency (Hz)	IHS (g)
0.1	0.013
0.3	0.067
1.2	0.301
2	0.502
2.2	0.551
8	0.551
10	0.490
12	0.445
27	0.289
33	0.260
100	0.260

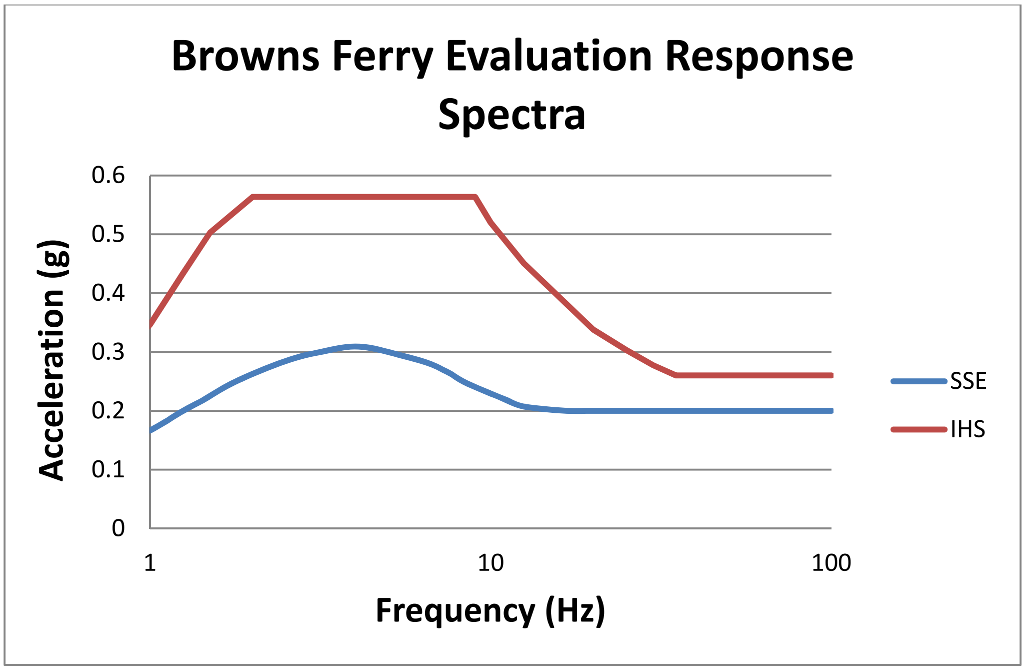


Figure 3.3-1. SSE and IHS Response Spectra for Browns Ferry Nuclear Plant.

4.0 Screening Evaluation

In accordance with SPID (EPRI, 2013a) Section 3, a screening evaluation was performed as described below.

4.1 Risk Evaluation Screening (1 to 10 Hz)

In the 1 to 10 Hz part of the response spectrum, the IHS exceeds the GMRS. Based on this comparison, a risk evaluation will not be performed.

4.2 High Frequency Screening (> 10 Hz)

For the range above 10 Hz, the GMRS exceeds the SSE. Therefore, Browns Ferry Nuclear Plant screens in for a High Frequency Confirmation.

4.3 Spent Fuel Pool Evaluation Screening (1 to 10 Hz)

In the 1 to 10 Hz part of the response spectrum, the GMRS exceeds the SSE. Therefore, Browns Ferry Nuclear Plant screens in for a Spent Fuel Pool evaluation.

5.0 Interim Actions

Based on the screening evaluation, the expedited seismic evaluation described in EPRI 3002000704 (EPRI, 2013c) will be performed as proposed in a letter to NRC (ML13101A379) dated April 9, 2013 (NEI, 2013) and agreed to by NRC (ML13106A331) in a letter dated May 7, 2013 (U.S. NRC, 2013).

As part of the ESEP process, the Reactor Core Isolation Cooling will be included in the single success path strategy. This will resolve recommendations from the Staff Evaluation Report of the IPEEEs (U.S. NRC, 2000).

Consistent with NRC letter (ML14030A046) dated February 20, 2014, (U.S. NRC, 2014a) the seismic hazard reevaluations presented herein are distinct from the current design and licensing bases of Browns Ferry Nuclear Plant. Therefore, the results do not call into question the operability or functionality of Structures, Systems, and Components (SSCs) and are not reportable pursuant to 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors," and 10 CFR 50.73, "Licensee event report system."

The NRC letter also requests that licensees provide an interim evaluation or actions to demonstrate that the plant can cope with the reevaluated hazard while the expedited approach and risk evaluations are conducted. In response to that request, Nuclear Energy Institute (NEI) letter dated March 12, 2014 (NEI, 2014), provides seismic core damage risk estimates using the updated seismic hazards for the operating nuclear plants in the Central and Eastern United

States. These risk estimates continue to support the following conclusions of the NRC GI-199 Safety/Risk Assessment (U.S. NRC, 2010):

> Overall seismic core damage risk estimates are consistent with the Commission's Safety Goal Policy Statement because they are within the subsidiary objective of 10^{-4}/year for core damage frequency. The GI-199 Safety/Risk Assessment, based in part on information from the U.S. NRC's Individual Plant Examination of External Events (IPEEE) program, indicates that no concern exists regarding adequate protection and that the current seismic design of operating reactors provides a safety margin to withstand potential earthquakes exceeding the original design basis.

Browns Ferry Nuclear Plant is included in the March 12, 2014 risk estimates (NEI, 2014). Using the methodology described in the NEI letter, all plants were shown to be below 10^{-4}/year; thus, the above conclusions apply.

In accordance with the Near-Term Task Force Recommendation 2.3 Seismic (U.S. NRC, 2012) Browns Ferry Nuclear Plant Units 1, 2, and 3 performed seismic walkdowns using the guidance in EPRI Report 1025286 (EPRI, 2012). The seismic walkdowns were completed and captured in the seismic walkdown reports (TVA, 2012) (TVA, 2013). At Browns Ferry Nuclear Plant Units 1, 2 and 3 a total of 120 equipment items per unit were selected from the IPEEE Safe Shutdown Equipment List (SSEL) to fulfill the requirements of the seismic walkdown guidance (TVA, 2012). The selected items were located in various environments and included many different types of equipment from multiple safety systems.

Two potentially adverse seismic conditions were identified for Unit 1 (TVA, 2012), three potentially adverse seismic conditions were identified for Unit 2 (TVA, 2013), and no potentially adverse seismic conditions were identified for Unit 3 (TVA, 2012) and entered into the TVA Corrective Action Program. The identified potentially adverse conditions were evaluated and were found to have no operability or reportability impact on the plant. All potentially adverse seismic conditions identified for Browns Ferry Nuclear Plant Units 1 and 2 have been resolved. Based on the NRC Staff's review of the seismic walkdown reports, the NRC Staff concluded that Browns Ferry Nuclear Plant's implementation of the seismic walkdown methodology meets the intent of the walkdown guidance and that no immediate safety concerns were identified (U.S. NRC, 2014b, U.S. NRC 2014c, U.S. NRC, 2014d).

The seismic walkdowns also verified in Section 7.0 that any IPEEE outliers or vulnerabilities identified were adequately addressed. Resolutions to all of the outliers or vulnerabilities have been identified during the IPEEE program as described below in Table 5.0-1. The only Browns Ferry Nuclear Plant IPEEE vulnerabilities were the Unit 1 and 2 common Diesel Auxiliary Board Transformers (TDA and TDB) which were found to have a HCLPF capacity of 0.26g. These are being resolved by replacements as described in Table 5.0-1. Transformer TDA was replaced in January 2014 and transformer TDB is scheduled to be replaced by September 30, 2014 (TVA, 2012). These transformers were the only Browns Ferry Nuclear Plant IPEEE vulnerabilities because the USI A-46 efforts identified and resolved Generic Implementation Procedure (GIP) outliers and other seismic performance concerns, by plant modification and work orders, ahead of the IPEEE walkdowns.

Table 5.0-1. IPEEE Outliers. (TVA, 2012)

Equipment Name	Resolution
Diesel Auxiliary Board Transformer (TDA)	Transformer has been replaced
Diesel Auxiliary Board Transformer (TDB)	Transformer replacement

6.0 Conclusions

In accordance with the 50.54(f) request for information (U.S. NRC, 2012a), a seismic hazard and screening evaluation was performed for Browns Ferry Nuclear Plant. A GMRS was developed solely for purpose of screening for additional evaluations in accordance with the SPID (EPRI, 2013a). Based on the results of the screening evaluation, Browns Ferry Nuclear Plant screens-in for a Spent Fuel Pool evaluation and a High Frequency Confirmation. Additionally, based on the results of the screening evaluation, Browns Ferry Nuclear Plant screens out for a seismic risk evaluation.

7.0 References

10 CFR Part 50. Title 10, Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington DC.

10 CFR Part 100. Title 10, Code of Federal Regulations, Part 100, "Reactor Site Criteria," U.S. Nuclear Regulatory Commission, Washington DC.

10 CFR Part 50.72. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, Washington DC.

10 CFR Part 50.73. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System," U.S. Nuclear Regulatory Commission, Washington DC.

AMEC (2013). "Seismic Data Retrieval Information for EPRI Near-Term Task Force Recommendation 2.1 TVA Browns Ferry Nuclear Plant Athens, Alabama," AMEC Project 3043121013 report transmitted by letter from K. Campbell to J. Best, June 26, 2013.

CEUS-SSC (2012). "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," U.S. Nuclear Regulatory Commission Report, NUREG-2115; EPRI Report 1021097, 6 Volumes; DOE Report# DOE/NE-0140.

EPRI (1991). "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," EPRI NP-6041-SL, Revision 1, August 1991.

EPRI (2012). "Seismic Walkdown Guidance: For Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic," Electric Power Research Institute, Report 1025286, June 4, 2012.

EPRI (2013a). "Seismic Evaluation Guidance Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," Electric Power Research Institute Report 1025287, February 2013.

EPRI (2013b). "EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project," Electric Power Research Institute, Palo Alto, CA, Rept. 3002000717, 2 volumes, June 2013.

EPRI (2013c). EPRI 3002000704, "Seismic Evaluation Guidance, Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," May 2013.

EPRI (2014). "Browns Ferry Seismic Hazard and Screening Report," Electric Power Research Institute, Palo Alto, CA, dated February 28, 2014.

GSA (1969). “Stratigraphic Succession along the Appalachian Structural Front in Alabama,” James A. Drahovzal and D.E. Raymond Thornton L. Neathery, Abstracts with Programs, part. 7, page 158.

Lemiszki, P.J., Kohl, M.S., and Sutton, E.F. (2008). Geologic Map and Mineral Resources Summary of the Decatur Quadrangle: Tennessee Division of Geology, Geologic Quadrangle Map 118 SE, scale 1:24,000.”

NEI (2013). NEI Letter to NRC, “Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations,” April 9, 2013.

NEI (2014). NEI Letter from A. Pietrangelo to E. Leeds, “Seismic Risk Evaluations for Plants in the Central and Eastern United States,” March 12, 2014.

Toro (1997). Appendix of: Silva, W.J., Abrahamson, N., Toro, G., and Costantino, C. (1997). “Description and Validation of the Stochastic Ground Motion Model,” Report Submitted to Brookhaven National Laboratory, Associated Universities, Inc., Upton, New York 11973, Contract No. 770573.

TVA (1996). TVA Letter from P. Salas to NRC, “Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Generic Letter (GL 87-02, Supplement 1, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue A-46 and GL 88-20, Supplement 4, individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Submittal of Seismic Evaluation Reports (TAC Nos. M69431, M69432, M83596 and M83589),” June 28, 1996.

TVA (2005). TVA Letter from T. E. Abney to NRC, “Browns Ferry Nuclear Plant (BFN) - Unit 1 Response to NRC GL 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Submittal of Browns Ferry Nuclear Plant Unit 1 Seismic and Internal Fires IPEEE,” January 14, 2005.

TVA (2012). Letter from J. Shea to NRC, “Tennessee Valley Authority (TVA) – Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding the Browns Ferry Nuclear Plant Seismic Walkdown Results of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” November 27, 2012.

TVA (2013). Letter from J. Shea to NRC, “Tennessee Valley Authority - Supplemental Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding the Browns Ferry Nuclear Plant, Unit 2 Seismic Walkdown Results of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” June 28, 2013.

TVA (Amendment 25.3). “Browns Ferry Nuclear Power Station – Final Safety Analysis Report,” Amendment 25.3.

U.S. NRC (1978). “Development of Criteria for Seismic Review of Selected Nuclear Power Plants,” NUREG/CR-0098, May 1978.

U.S. NRC (2000). Letter from NRC to J. A. Scalice, “Browns Ferry Units 1, 2 and 3 , Individual Plant Examination of External Events (IPEEE) and Related Generic Safety Issues, Issuance of Staff Evaluation (TAC Nos M83595, M83596, M83679),” June 22, 2000.

U.S. NRC (2007a). “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” U.S. Nuclear Regulatory Commission Reg. Guide 1.208, March 2007.

U.S. NRC (2007b). Letter to William R. Campbell from NRC, “Browns Ferry Nuclear Plant, Unit 1 - Closeout of Generic Letter 88-20, Supplement 4, Concerning Individual Plant

Examination of External Events for Severe Accident Vulnerabilities (TAC No. MC5729)" June 28, 2007.

U.S. NRC (2010). "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," GI-199, September 2, 2010.

U.S. NRC (2012). NRC (E Leeds and M Johnson) Letter to All Power Reactor Licensees et al., "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," March 12, 2012.

U.S. NRC (2013). NRC Letter, Eric J. Leeds to Joseph E. Pollock, NEI "Electric Power Research Institute Final Draft Report XXXXXX, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, As an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluation," May 7, 2013.

U.S. NRC (2014a). NRC Letter, Eric J. Leeds to All Power Reactor Licensees, "Supplemental Information Related to Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," February 20, 2014.

U.S. NRC (2014b). NRC Letter, F. Saba to J. Shea, "Browns Ferry Nuclear Plant, Unit 1 - Staff Assessment of the Seismic Walkdown Report Supporting Implementation of Near-Term Task Force Recommendation 2.3 Related to the Fukushima Dai-ichi Nuclear Power Plant Accident (TAC No. MF0096)," March 18, 2014.

U.S. NRC (2014c). NRC Letter, F. Saba to J. Shea, "Browns Ferry Nuclear Plant, Unit 2 - Staff Assessment of the Seismic Walkdown Report Supporting Implementation of Near-Term Task Force Recommendation 2.3 Related to the Fukushima Dai-ichi Nuclear Power Plant Accident (TAC No. MF0097)," March 18, 2014.

U.S. NRC (2014d). NRC Letter, F. Saba to J. Shea, "Browns Ferry Nuclear Plant, Unit 3 - Staff Assessment of the Seismic Walkdown Report Supporting Implementation of Near-Term Task Force Recommendation 2.3 Related to the Fukushima Dai-ichi Nuclear Power Plant Accident (TAC No. MF0098)," March 18, 2014.

Appendix A

Tabulated Data

Table A-1a. Mean and Fractile Seismic Hazard Curves for 0.5 Hz at Browns Ferry Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	4.11E-02	2.01E-02	3.01E-02	4.01E-02	5.20E-02	6.17E-02
0.001	2.40E-02	1.10E-02	1.67E-02	2.29E-02	3.14E-02	4.01E-02
0.005	5.88E-03	1.98E-03	3.09E-03	5.42E-03	8.72E-03	1.15E-02
0.01	3.09E-03	6.36E-04	1.20E-03	2.76E-03	4.98E-03	6.73E-03
0.015	2.05E-03	2.68E-04	5.83E-04	1.67E-03	3.52E-03	5.12E-03
0.03	8.39E-04	4.31E-05	1.18E-04	5.05E-04	1.57E-03	2.76E-03
0.05	3.41E-04	8.47E-06	2.60E-05	1.44E-04	6.09E-04	1.36E-03
0.075	1.40E-04	2.07E-06	6.54E-06	4.19E-05	2.22E-04	6.17E-04
0.1	6.71E-05	7.03E-07	2.29E-06	1.57E-05	9.65E-05	3.05E-04
0.15	2.07E-05	1.40E-07	4.83E-07	3.47E-06	2.39E-05	9.51E-05
0.3	1.95E-06	5.91E-09	2.53E-08	2.13E-07	1.84E-06	8.12E-06
0.5	2.90E-07	5.05E-10	2.32E-09	2.42E-08	2.60E-07	1.25E-06
0.75	6.68E-08	1.79E-10	4.01E-10	3.95E-09	5.27E-08	3.05E-07
1.	2.55E-08	1.72E-10	1.92E-10	1.15E-09	1.67E-08	1.18E-07
1.5	7.23E-09	1.21E-10	1.72E-10	2.72E-10	3.37E-09	3.01E-08
3.	8.33E-10	1.11E-10	1.21E-10	1.72E-10	2.88E-10	2.42E-09
5.	1.44E-10	1.11E-10	1.21E-10	1.72E-10	1.72E-10	3.95E-10
7.5	3.09E-11	1.11E-10	1.11E-10	1.72E-10	1.72E-10	1.82E-10
10.	9.54E-12	1.11E-10	1.11E-10	1.72E-10	1.72E-10	1.72E-10

Table A-1b. Mean and Fractile Seismic Hazard Curves for 1.0 Hz at Browns Ferry Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	8.13E-02	4.37E-02	5.83E-02	8.23E-02	9.93E-02	9.93E-02
0.001	5.47E-02	2.42E-02	3.52E-02	5.50E-02	7.23E-02	8.60E-02
0.005	1.33E-02	5.05E-03	7.55E-03	1.23E-02	1.87E-02	2.42E-02
0.01	6.41E-03	2.13E-03	3.37E-03	5.91E-03	9.37E-03	1.23E-02
0.015	4.17E-03	1.11E-03	1.95E-03	3.79E-03	6.36E-03	8.47E-03
0.03	1.91E-03	2.72E-04	5.75E-04	1.55E-03	3.28E-03	4.77E-03
0.05	9.16E-04	7.13E-05	1.69E-04	6.09E-04	1.69E-03	2.80E-03
0.075	4.30E-04	2.07E-05	5.27E-05	2.25E-04	7.77E-04	1.55E-03
0.1	2.25E-04	8.00E-06	2.10E-05	9.79E-05	3.95E-04	8.72E-04
0.15	7.70E-05	1.92E-06	5.20E-06	2.53E-05	1.25E-04	3.23E-04
0.3	8.24E-06	1.27E-07	4.01E-07	2.07E-06	1.11E-05	3.52E-05
0.5	1.30E-06	1.29E-08	5.05E-08	3.14E-07	1.79E-06	5.35E-06
0.75	3.08E-07	1.77E-09	8.12E-09	6.73E-08	4.37E-07	1.32E-06
1.	1.19E-07	4.77E-10	2.13E-09	2.19E-08	1.64E-07	5.42E-07
1.5	3.37E-08	1.82E-10	3.90E-10	4.19E-09	4.13E-08	1.57E-07
3.	3.82E-09	1.23E-10	1.72E-10	2.92E-10	3.14E-09	1.62E-08
5.	6.49E-10	1.11E-10	1.21E-10	1.72E-10	4.56E-10	2.42E-09
7.5	1.38E-10	1.11E-10	1.21E-10	1.72E-10	1.84E-10	5.27E-10
10.	4.23E-11	1.11E-10	1.21E-10	1.72E-10	1.72E-10	2.35E-10

Table A-1c. Mean and Fractile Seismic Hazard Curves for 2.5 Hz at Browns Ferry Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.14E-01	8.98E-02	9.93E-02	9.93E-02	9.93E-02	9.93E-02
0.001	9.38E-02	6.36E-02	7.55E-02	9.37E-02	9.93E-02	9.93E-02
0.005	3.06E-02	1.44E-02	2.07E-02	2.88E-02	4.13E-02	4.98E-02
0.01	1.39E-02	6.45E-03	8.98E-03	1.32E-02	1.90E-02	2.35E-02
0.015	8.42E-03	3.79E-03	5.27E-03	8.00E-03	1.16E-02	1.44E-02
0.03	3.63E-03	1.20E-03	1.84E-03	3.37E-03	5.42E-03	7.13E-03
0.05	1.88E-03	3.95E-04	6.73E-04	1.55E-03	3.09E-03	4.50E-03
0.075	1.01E-03	1.40E-04	2.57E-04	6.83E-04	1.77E-03	3.01E-03
0.1	6.02E-04	6.26E-05	1.16E-04	3.33E-04	1.05E-03	2.07E-03
0.15	2.58E-04	1.82E-05	3.52E-05	1.05E-04	4.01E-04	1.08E-03
0.3	4.35E-05	1.87E-06	4.07E-06	1.21E-05	5.12E-05	1.87E-04
0.5	8.99E-06	3.19E-07	7.77E-07	2.68E-06	1.04E-05	3.28E-05
0.75	2.31E-06	6.83E-08	1.92E-07	8.12E-07	3.05E-06	7.45E-06
1.	8.62E-07	2.10E-08	6.73E-08	3.28E-07	1.29E-06	2.84E-06
1.5	2.19E-07	3.47E-09	1.29E-08	8.23E-08	3.57E-07	8.60E-07
3.	2.11E-08	2.35E-10	6.09E-10	5.05E-09	3.33E-08	9.51E-08
5.	3.23E-09	1.62E-10	1.74E-10	5.50E-10	4.25E-09	1.51E-08
7.5	6.29E-10	1.21E-10	1.62E-10	1.90E-10	7.55E-10	2.92E-09
10.	1.80E-10	1.11E-10	1.21E-10	1.72E-10	2.80E-10	8.85E-10

Table A-1d. Mean and Fractile Seismic Hazard Curves for 5.0 Hz at Browns Ferry Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.19E-01	9.79E-02	9.93E-02	9.93E-02	9.93E-02	9.93E-02
0.001	1.04E-01	7.45E-02	8.72E-02	9.93E-02	9.93E-02	9.93E-02
0.005	4.24E-02	1.98E-02	3.01E-02	4.13E-02	5.58E-02	6.36E-02
0.01	2.08E-02	9.11E-03	1.38E-02	2.01E-02	2.84E-02	3.37E-02
0.015	1.28E-02	5.75E-03	8.23E-03	1.23E-02	1.77E-02	2.13E-02
0.03	5.44E-03	2.16E-03	3.19E-03	5.12E-03	7.77E-03	9.65E-03
0.05	2.87E-03	8.35E-04	1.31E-03	2.53E-03	4.43E-03	6.09E-03
0.075	1.63E-03	3.37E-04	5.58E-04	1.27E-03	2.76E-03	4.13E-03
0.1	1.03E-03	1.67E-04	2.84E-04	6.93E-04	1.74E-03	3.09E-03
0.15	4.91E-04	5.91E-05	1.05E-04	2.64E-04	7.89E-04	1.84E-03
0.3	1.04E-04	9.79E-06	1.82E-05	4.43E-05	1.31E-04	4.37E-04
0.5	2.60E-05	2.49E-06	4.83E-06	1.23E-05	3.28E-05	9.37E-05
0.75	7.85E-06	7.66E-07	1.60E-06	4.37E-06	1.15E-05	2.46E-05
1.	3.31E-06	3.05E-07	6.73E-07	2.01E-06	5.20E-06	1.02E-05
1.5	9.74E-07	7.13E-08	1.77E-07	6.00E-07	1.64E-06	3.19E-06
3.	1.10E-07	3.68E-09	1.13E-08	5.35E-08	1.90E-07	4.13E-07
5.	1.82E-08	3.90E-10	1.07E-09	6.36E-09	2.96E-08	7.66E-08
7.5	3.76E-09	1.72E-10	2.42E-10	1.01E-09	5.50E-09	1.67E-08
10.	1.12E-09	1.34E-10	1.72E-10	3.19E-10	1.55E-09	5.20E-09

Table A-1e. Mean and Fractile Seismic Hazard Curves for 10 Hz at Browns Ferry Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.17E-01	9.65E-02	9.93E-02	9.93E-02	9.93E-02	9.93E-02
0.001	1.02E-01	7.45E-02	8.85E-02	9.93E-02	9.93E-02	9.93E-02
0.005	4.34E-02	2.19E-02	3.19E-02	4.31E-02	5.42E-02	6.83E-02
0.01	2.26E-02	1.05E-02	1.46E-02	2.19E-02	2.84E-02	4.07E-02
0.015	1.45E-02	6.64E-03	9.11E-03	1.38E-02	1.87E-02	2.84E-02
0.03	6.54E-03	2.60E-03	3.73E-03	5.91E-03	8.98E-03	1.34E-02
0.05	3.55E-03	1.08E-03	1.64E-03	3.05E-03	5.42E-03	7.89E-03
0.075	2.09E-03	4.70E-04	7.45E-04	1.62E-03	3.47E-03	5.27E-03
0.1	1.38E-03	2.46E-04	4.07E-04	9.51E-04	2.39E-03	3.95E-03
0.15	7.08E-04	9.65E-05	1.64E-04	4.01E-04	1.23E-03	2.39E-03
0.3	1.77E-04	1.98E-05	3.57E-05	8.85E-05	2.57E-04	6.36E-04
0.5	5.21E-05	6.26E-06	1.20E-05	2.88E-05	7.13E-05	1.67E-04
0.75	1.84E-05	2.35E-06	4.56E-06	1.15E-05	2.68E-05	5.50E-05
1.	8.62E-06	1.08E-06	2.19E-06	5.75E-06	1.34E-05	2.57E-05
1.5	2.92E-06	3.14E-07	6.83E-07	1.95E-06	4.90E-06	8.85E-06
3.	4.00E-07	2.29E-08	6.09E-08	2.29E-07	7.03E-07	1.36E-06
5.	7.60E-08	2.16E-09	6.83E-09	3.37E-08	1.32E-07	2.92E-07
7.5	1.75E-08	3.57E-10	1.01E-09	6.00E-09	2.92E-08	7.34E-08
10.	5.65E-09	1.84E-10	3.09E-10	1.57E-09	8.98E-09	2.49E-08

Table A-1f. Mean and Fractile Seismic Hazard Curves for 25 Hz at Browns Ferry Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.11E-01	7.66E-02	9.79E-02	9.93E-02	9.93E-02	9.93E-02
0.001	9.28E-02	5.35E-02	8.00E-02	9.37E-02	9.93E-02	9.93E-02
0.005	3.61E-02	1.55E-02	2.60E-02	3.42E-02	4.37E-02	7.03E-02
0.01	1.91E-02	8.00E-03	1.21E-02	1.74E-02	2.32E-02	4.50E-02
0.015	1.26E-02	5.20E-03	7.55E-03	1.11E-02	1.57E-02	3.23E-02
0.03	6.03E-03	2.04E-03	2.96E-03	5.12E-03	8.23E-03	1.57E-02
0.05	3.35E-03	8.12E-04	1.27E-03	2.64E-03	5.27E-03	8.85E-03
0.075	2.03E-03	3.57E-04	5.83E-04	1.38E-03	3.52E-03	6.00E-03
0.1	1.38E-03	1.98E-04	3.28E-04	8.23E-04	2.49E-03	4.43E-03
0.15	7.58E-04	8.72E-05	1.51E-04	3.84E-04	1.34E-03	2.80E-03
0.3	2.24E-04	2.35E-05	4.31E-05	1.05E-04	3.14E-04	8.60E-04
0.5	7.60E-05	8.72E-06	1.64E-05	4.07E-05	1.02E-04	2.49E-04
0.75	2.97E-05	3.47E-06	6.93E-06	1.77E-05	4.31E-05	8.60E-05
1.	1.49E-05	1.64E-06	3.47E-06	9.37E-06	2.32E-05	4.25E-05
1.5	5.53E-06	4.77E-07	1.11E-06	3.52E-06	9.37E-06	1.69E-05
3.	9.06E-07	3.09E-08	9.65E-08	4.83E-07	1.67E-06	3.19E-06
5.	2.01E-07	2.39E-09	1.01E-08	8.12E-08	3.73E-07	8.00E-07
7.5	5.33E-08	3.28E-10	1.31E-09	1.60E-08	9.51E-08	2.29E-07
10.	1.91E-08	1.74E-10	3.52E-10	4.43E-09	3.23E-08	8.60E-08

Table A-1g. Mean and Fractile Seismic Hazard Curves for 100 Hz (PGA) at Browns Ferry Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.02E-01	5.75E-02	8.72E-02	9.93E-02	9.93E-02	9.93E-02
0.001	7.88E-02	3.52E-02	6.45E-02	7.89E-02	9.65E-02	9.93E-02
0.005	2.24E-02	8.47E-03	1.40E-02	2.10E-02	2.76E-02	5.12E-02
0.01	1.06E-02	4.01E-03	5.91E-03	9.24E-03	1.32E-02	2.92E-02
0.015	6.73E-03	2.22E-03	3.33E-03	5.66E-03	8.85E-03	1.92E-02
0.03	2.95E-03	5.91E-04	9.65E-04	2.22E-03	4.70E-03	8.60E-03
0.05	1.46E-03	1.87E-04	3.19E-04	8.47E-04	2.60E-03	5.05E-03
0.075	7.68E-04	7.66E-05	1.32E-04	3.52E-04	1.36E-03	3.01E-03
0.1	4.62E-04	4.25E-05	7.45E-05	1.92E-04	7.34E-04	1.90E-03
0.15	2.09E-04	1.90E-05	3.52E-05	8.60E-05	2.72E-04	8.60E-04
0.3	4.37E-05	4.50E-06	8.72E-06	2.22E-05	5.66E-05	1.53E-04
0.5	1.26E-05	1.21E-06	2.60E-06	7.34E-06	1.90E-05	4.01E-05
0.75	4.63E-06	3.33E-07	8.12E-07	2.68E-06	7.55E-06	1.49E-05
1.	2.21E-06	1.13E-07	3.19E-07	1.21E-06	3.79E-06	7.45E-06
1.5	7.29E-07	1.90E-08	6.64E-08	3.33E-07	1.25E-06	2.68E-06
3.	8.33E-08	5.12E-10	2.46E-09	2.25E-08	1.34E-07	3.57E-07
5.	1.27E-08	1.72E-10	2.53E-10	2.07E-09	1.79E-08	5.91E-08
7.5	2.32E-09	1.21E-10	1.72E-10	3.42E-10	2.84E-09	1.13E-08
10.	6.17E-10	1.11E-10	1.32E-10	1.79E-10	7.45E-10	3.05E-09

Table A-2. Amplification Functions for Browns Ferry Nuclear Plant. (EPRI, 2014)

PGA	Median AF	Sigma ln(AF)	25 Hz	Median AF	Sigma ln(AF)	10 Hz	Median AF	Sigma ln(AF)	5.0 Hz	Median AF	Sigma ln(AF)
1.00E-02	1.04E+00	3.95E-02	1.30E-02	1.00E+00	4.82E-02	1.90E-02	9.67E-01	9.02E-02	2.09E-02	1.03E+00	6.78E-02
4.95E-02	9.63E-01	5.10E-02	1.02E-01	9.40E-01	6.65E-02	9.99E-02	9.41E-01	1.06E-01	8.24E-02	1.02E+00	7.07E-02
9.64E-02	9.39E-01	5.47E-02	2.13E-01	9.27E-01	6.98E-02	1.85E-01	9.34E-01	1.07E-01	1.44E-01	1.01E+00	7.10E-02
1.94E-01	9.17E-01	5.81E-02	4.43E-01	9.11E-01	7.28E-02	3.56E-01	9.24E-01	1.08E-01	2.65E-01	1.01E+00	7.09E-02
2.92E-01	9.03E-01	6.05E-02	6.76E-01	8.99E-01	7.55E-02	5.23E-01	9.17E-01	1.08E-01	3.84E-01	1.00E+00	7.12E-02
3.91E-01	8.92E-01	6.23E-02	9.09E-01	8.89E-01	7.77E-02	6.90E-01	9.11E-01	1.09E-01	5.02E-01	1.00E+00	7.14E-02
4.93E-01	8.84E-01	6.37E-02	1.15E+00	8.80E-01	7.95E-02	8.61E-01	9.06E-01	1.09E-01	6.22E-01	9.97E-01	7.14E-02
7.41E-01	8.66E-01	6.56E-02	1.73E+00	8.60E-01	8.22E-02	1.27E+00	8.95E-01	1.10E-01	9.13E-01	9.91E-01	7.16E-02
1.01E+00	8.52E-01	6.62E-02	2.36E+00	8.43E-01	8.37E-02	1.72E+00	8.85E-01	1.10E-01	1.22E+00	9.85E-01	7.12E-02
1.28E+00	8.40E-01	6.61E-02	3.01E+00	8.29E-01	8.40E-02	2.17E+00	8.76E-01	1.10E-01	1.54E+00	9.80E-01	7.11E-02
1.55E+00	8.31E-01	6.55E-02	3.63E+00	8.17E-01	8.38E-02	2.61E+00	8.68E-01	1.10E-01	1.85E+00	9.75E-01	7.08E-02
2.5 Hz	**Median AF**	**Sigma ln(AF)**	**1.0 Hz**	**Median AF**	**Sigma ln(AF)**	**0.5 Hz**	**Median AF**	**Sigma ln(AF)**			
2.18E-02	1.06E+00	6.22E-02	1.27E-02	1.24E+00	1.32E-01	8.25E-03	1.20E+00	1.03E-01			
7.05E-02	1.05E+00	6.19E-02	3.43E-02	1.23E+00	1.28E-01	1.96E-02	1.19E+00	9.99E-02			
1.18E-01	1.05E+00	6.16E-02	5.51E-02	1.23E+00	1.26E-01	3.02E-02	1.19E+00	9.88E-02			
2.12E-01	1.05E+00	6.14E-02	9.63E-02	1.23E+00	1.25E-01	5.11E-02	1.19E+00	9.80E-02			
3.04E-01	1.04E+00	6.12E-02	1.36E-01	1.23E+00	1.24E-01	7.10E-02	1.19E+00	9.77E-02			
3.94E-01	1.04E+00	6.10E-02	1.75E-01	1.23E+00	1.24E-01	9.06E-02	1.19E+00	9.75E-02			
4.86E-01	1.04E+00	6.10E-02	2.14E-01	1.23E+00	1.23E-01	1.10E-01	1.19E+00	9.75E-02			
7.09E-01	1.04E+00	6.09E-02	3.10E-01	1.23E+00	1.23E-01	1.58E-01	1.19E+00	9.75E-02			
9.47E-01	1.04E+00	6.06E-02	4.12E-01	1.23E+00	1.22E-01	2.09E-01	1.19E+00	9.76E-02			
1.19E+00	1.04E+00	6.05E-02	5.18E-01	1.23E+00	1.22E-01	2.62E-01	1.19E+00	9.76E-02			
1.43E+00	1.03E+00	6.07E-02	6.19E-01	1.23E+00	1.22E-01	3.12E-01	1.19E+00	9.78E-02			

Tables A-3a and A-3b are tabular versions of the typical amplification factors provided in Figures 2.3.6-1 and 2.3.6-2. Values are provided for two input motion levels at approximately 10^{-4} and 10^{-5} mean annual frequency of exceedance. These factors are unverified and are provided for information only. The figures should be considered the governing information.

Table A-3a. Median AFs and sigmas for Model 1, Profile 1, for 2 PGA levels.

For Information Only

M1P1K1	Rock PGA=0.194			M1P1K1	PGA=0.741		
Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)	Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)
100.0	0.180	0.929	0.075	100.0	0.607	0.819	0.099
87.1	0.185	0.928	0.078	87.1	0.621	0.812	0.103
75.9	0.193	0.925	0.082	75.9	0.648	0.800	0.111
66.1	0.209	0.923	0.091	66.1	0.704	0.779	0.127
57.5	0.241	0.910	0.101	57.5	0.811	0.745	0.150
50.1	0.290	0.911	0.121	50.1	0.971	0.731	0.167
43.7	0.339	0.899	0.143	43.7	1.136	0.724	0.185
38.0	0.362	0.873	0.123	38.0	1.223	0.718	0.174
33.1	0.386	0.879	0.107	33.1	1.299	0.733	0.160
28.8	0.417	0.949	0.087	28.8	1.402	0.803	0.138
25.1	0.435	0.982	0.087	25.1	1.467	0.846	0.116
21.9	0.425	1.006	0.132	21.9	1.442	0.887	0.144
19.1	0.396	0.950	0.143	19.1	1.340	0.848	0.158
16.6	0.360	0.897	0.154	16.6	1.212	0.810	0.167
14.5	0.332	0.865	0.123	14.5	1.113	0.788	0.136
12.6	0.319	0.854	0.118	12.6	1.063	0.782	0.128
11.0	0.306	0.840	0.097	11.0	1.031	0.785	0.107
9.5	0.315	0.905	0.103	9.5	1.048	0.843	0.112
8.3	0.293	0.914	0.090	8.3	0.987	0.869	0.093
7.2	0.280	0.931	0.073	7.2	0.925	0.877	0.079
6.3	0.295	1.044	0.080	6.3	0.959	0.975	0.093
5.5	0.289	1.069	0.073	5.5	0.952	1.021	0.071
4.8	0.262	0.991	0.073	4.8	0.878	0.968	0.070
4.2	0.247	0.964	0.045	4.2	0.828	0.947	0.053
3.6	0.242	0.969	0.049	3.6	0.807	0.954	0.053
3.2	0.248	1.055	0.050	3.2	0.824	1.040	0.052
2.8	0.239	1.073	0.055	2.8	0.797	1.066	0.055
2.4	0.227	1.104	0.083	2.4	0.759	1.105	0.083
2.1	0.207	1.105	0.091	2.1	0.691	1.112	0.088
1.8	0.188	1.126	0.075	1.8	0.627	1.134	0.074
1.6	0.168	1.158	0.098	1.6	0.557	1.166	0.098
1.4	0.155	1.238	0.144	1.4	0.508	1.243	0.142
1.2	0.139	1.266	0.132	1.2	0.455	1.270	0.132
1.0	0.122	1.230	0.135	1.0	0.396	1.234	0.133
0.91	0.106	1.176	0.096	0.91	0.342	1.179	0.095
0.79	0.096	1.174	0.095	0.79	0.306	1.176	0.093
0.69	0.087	1.193	0.108	0.69	0.275	1.194	0.107
0.60	0.076	1.196	0.094	0.60	0.238	1.197	0.093
0.52	0.066	1.218	0.086	0.52	0.205	1.217	0.085
0.46	0.058	1.286	0.099	0.46	0.179	1.286	0.100
0.10	0.002	1.208	0.046	0.10	0.007	1.198	0.049

Table A-3b. Median AFs and sigmas for Model 2, Profile 1, for 2 PGA levels.

For Information Only

M2P1K1		PGA=0.194		M2P1K1		PGA=0.741	
Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)	Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)
100.0	0.187	0.964	0.049	100.0	0.710	0.958	0.050
87.1	0.192	0.964	0.048	87.1	0.734	0.960	0.050
75.9	0.201	0.965	0.049	75.9	0.779	0.962	0.050
66.1	0.220	0.969	0.047	66.1	0.876	0.970	0.049
57.5	0.256	0.966	0.044	57.5	1.052	0.966	0.046
50.1	0.311	0.975	0.069	50.1	1.299	0.978	0.074
43.7	0.362	0.960	0.093	43.7	1.508	0.960	0.099
38.0	0.385	0.929	0.070	38.0	1.582	0.929	0.073
33.1	0.410	0.933	0.059	33.1	1.654	0.933	0.062
28.8	0.441	1.002	0.047	28.8	1.752	1.003	0.049
25.1	0.457	1.031	0.061	25.1	1.789	1.032	0.061
21.9	0.443	1.050	0.116	21.9	1.709	1.051	0.117
19.1	0.412	0.987	0.125	19.1	1.561	0.988	0.126
16.6	0.373	0.931	0.142	16.6	1.394	0.931	0.144
14.5	0.343	0.894	0.110	14.5	1.263	0.894	0.112
12.6	0.329	0.881	0.111	12.6	1.196	0.880	0.113
11.0	0.313	0.860	0.088	11.0	1.126	0.857	0.089
9.5	0.322	0.927	0.098	9.5	1.150	0.925	0.099
8.3	0.299	0.932	0.086	8.3	1.056	0.930	0.087
7.2	0.286	0.950	0.068	7.2	1.000	0.948	0.069
6.3	0.302	1.069	0.069	6.3	1.050	1.067	0.070
5.5	0.294	1.088	0.068	5.5	1.014	1.087	0.068
4.8	0.265	1.003	0.073	4.8	0.908	1.002	0.073
4.2	0.249	0.972	0.041	4.2	0.848	0.971	0.042
3.6	0.244	0.976	0.042	3.6	0.824	0.975	0.042
3.2	0.250	1.062	0.048	3.2	0.840	1.061	0.048
2.8	0.240	1.078	0.054	2.8	0.804	1.076	0.054
2.4	0.228	1.107	0.081	2.4	0.759	1.105	0.080
2.1	0.207	1.106	0.092	2.1	0.686	1.104	0.092
1.8	0.188	1.126	0.074	1.8	0.622	1.124	0.073
1.6	0.168	1.158	0.098	1.6	0.552	1.156	0.097
1.4	0.155	1.237	0.144	1.4	0.504	1.234	0.143
1.2	0.139	1.265	0.132	1.2	0.452	1.262	0.131
1.0	0.122	1.229	0.135	1.0	0.394	1.226	0.133
0.91	0.106	1.176	0.096	0.91	0.341	1.174	0.095
0.79	0.096	1.174	0.095	0.79	0.305	1.172	0.094
0.69	0.087	1.193	0.108	0.69	0.274	1.191	0.106
0.60	0.076	1.197	0.094	0.60	0.238	1.194	0.093
0.52	0.066	1.218	0.086	0.52	0.204	1.216	0.085
0.46	0.058	1.286	0.099	0.46	0.179	1.284	0.100
0.10	0.002	1.208	0.046	0.10	0.007	1.198	0.049

Appendix B

IPEEE Adequacy Review

Table of Contents

1.0 Background

The Nuclear Regulatory Commission (NRC) staff issued Generic Letter (GL) 88-20, Supplement 4 on June 28, 1991 (Reference 6.15), requesting that each licensee conduct an Individual Plant Examination of External Events (IPEEE) for severe accident vulnerabilities. Concurrently, NUREG-1407, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” was issued to provide utilities with detailed guidance for performance of the IPEEE (Reference 6.4).

A Seismic Margin Assessment (SMA) was performed for the seismic portion of Browns Ferry Nuclear Plant Units 1, 2 and 3 IPEEE using the EPRI SMA methodology, EPRI NP-6041-SL (Reference *6.3*) with enhancements identified in NUREG-1407 (Reference 6.4). Browns Ferry Nuclear Plant Units 1, 2 and 3 performed a 0.3g focused scope SMA utilizing a median centered NUREG/CR-0098 (Reference 6.11) spectral shape for a rock site. The calculated plant-level High Confidence of Low Probability of Failure (HCLPF) for Browns Ferry Nuclear Plant Unit 1 and 3 resulting from performance of the IPEEE was 0.3g and the calculated HCLPF for Unit 2 was reported as 0.26g. The results of the Browns Ferry Nuclear Plant Units 2 and 3 IPEEE were provided to NRC in a letter dated June 28, 1996 (Reference 6.1). The results of the BFN Unit 1 IPEEE were provided to NRC in a letter dated January 14, 2005 (Reference 6.15).

The NRC issued its Staff Evaluation Report (SER) on June 22, 2000 for the BFN Units 2 and 3 IPEEE (Reference 6.5) and subsequently on June 28, 2007 for the BFN Unit1 IPEEE (Reference 6.6). The SERs concluded that the BFN Units 1, 2 and 3 IPEEE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities, meeting the intent of GL 88-20 (Reference 6.15).

2.0 General Considerations

Browns Ferry Nuclear Plant located in Limestone County, Alabama was originally licensed for initial power under section 104(b) of the Atomic Energy Act of 1954, as amended, and the regulations of the Atomic Energy Commission set forth in Part 50 of Title 10 of the Code of Federal Regulations (10CFR50). Browns Ferry was not originally licensed under Appendix A of Part 100 - "Seismic and Geologic Siting Criteria for Nuclear Power Plants". However a review of Chapter 2.5 of Browns Ferry Nuclear Plant FSAR reveals that the elements presented in the FSAR meet the general expectations for investigations required to obtain the geologic and seismic data necessary to determine site suitability and provided reasonable assurance that a nuclear power plant can be constructed and operated at the proposed site without undue risk to the health and safety of the public.

Browns Ferry Nuclear Plant Units 1, 2 and 3 original licensing seismic Design Basis Earthquake (DBE), also referred to as Safe Shutdown Earthquake (SSE), is a Housner type spectrum anchored to 0.2g Peak Ground Acceleration (PGA). As part of Browns Ferry Nuclear Plant Units 1, 2 and 3 seismic reevaluation program, conducted 1986 thru 1989, updated dynamic analysis models of the safety-related structures were developed and submitted to NRC. The NRC Staff issued their Safety Evaluation Report of the updated models in NUREG-1232, Volume 3, Supplement 1 (Reference 6.12). The upgraded design response spectra were used for the Unresolved Safety Issue (USI) A-46 evaluation as well as for the design re-validation of all Browns Ferry Nuclear Plant safety related structures, systems, and components.

Browns Ferry Nuclear Plant Unit 1, 2 and 3 IPEEE is a focused scope EPRI seismic margin assessment. The IPEEE HCLPF Spectrum (IHS) is defined as the NUREG/CR-0098 median spectral shape for rock, anchored at the 0.3g PGA (Reference 6.11). The Review Level Earthquake (RLE) In-structure Response Spectra (ISRS) were developed by scaling up the upgraded design basis in-structure spectra based on scale factors. The dominant mode scaling procedure described in EPRI NP-6041-SL procedure was used to develop scale factors for Browns Ferry Nuclear Plant Units 1, 2 and 3 since the input motion for the A-46 and SMA earthquakes have similar shapes over the relevant range of frequencies (Reference 6.3).

The EPRI SMA method was selected as the method for the IPEEE evaluation because it was compatible with the Unresolved Safety Issue (USI) A-46 assessment being conducted in parallel with the IPEEE work. The Seismic Qualification Utility Group (SQUG) GIP (Reference 6.7) used for USI A-46 allowed for coordination of activities to support both projects. The Conservative Deterministic Failure Method (CDFM) was used to calculate the HCLPF capacities of components that did not screen out of evaluation by the SQUG GIP and EPRI SMA assessment. The results of these evaluations determined that the plant HCLPF capacity was greater than the RLE of 0.3g for Browns Ferry Nuclear Plant Units 1 and 3 and 0.26g for Unit 2. However, for IPEEE screening purposes at Browns Ferry Nuclear Plant Units 1, 2 and 3, the NUREG/CR-0098 IHS spectrum anchored at the RLE of 0.26g will be utilized for this review.

The IPEEE commitments and modifications that were required to achieve the plant level HCLPF capacity of 0.26g have been completed. Verification of the completion of these commitments and modifications were provided in the Browns Ferry Nuclear Plant Units 1, 2 and 3 Response to 10 CFR Part 50.54(f) (Enclosure 3) Request for Information Recommendation 2.3 Seismic (Reference 6.8) and are further discussed below in Section 3.0 Prerequisites. Confirmation that these modifications are still in place is described in the Prerequisites section of this report.

Since Browns Ferry Nuclear Plant was a focused scope IPEEE it is required to be upgraded to a full scope IPEEE for screening purposes according to the guidance in Section 3.3 of the SPID (Reference 6.18). The following sections summarize the results of the evaluation to upgrade to a full scope IPEEE.

2.1 Relay Chatter

The Browns Ferry Nuclear Plant Units 1, 2 and 3 relay evaluation for IPEEE was consistent with the requirements of a focused-scope evaluation, as described in NUREG-1407 (Reference 6.4). The full scope detailed review of relay chatter required in SPID Section 3.3.1 (Reference 6.18) has not been completed. As identified in the NEI letter to NRC dated October 3, 2013 (Reference 6.9), the relay chatter review will be completed on the same schedule as the High Frequency Confirmation as proposed in the NEI letter to NRC dated April 9, 2013 (Reference 6.19) and accepted in NRC's response dated May 7, 2013 (Reference 6.20).

2.2 Soil Failure Evaluation

As noted in NUREG-1407 Section 3.2.1, "a plant in full-scope category that is located on a rock site will not perform any soil failure evaluation." Browns Ferry Nuclear Plant is essentially a rock site with a shear wave velocity of 9500 feet/sec (Reference 6.22). The safety-related structures at Browns Ferry Nuclear Plant Units 1, 2 and 3, with the exception of the diesel generator building are founded on rock.

The diesel generator building is founded on engineered backfill of crushed stone. The crushed stone backfill was vibratory compacted that results in a very competent material that is not susceptible to liquefaction.

Regarding the in-situ soil at Browns Ferry Nuclear Plant, FSAR Chapter 12 (Reference 6.21), states in part "the soil structure is such that liquefaction should not be a problem". Additionally, it should be noted that based upon (Reference 6.22), underlying residual soils are fat clays and plastic silts which would be identified as CH and MH by the Unified Soil Classification System. As noted by "Liquefaction Susceptibility Criteria for Silts and Clays" (Reference 6.13). These types of soils are not sensitive to liquefaction effects. This further verifies the conclusion that soil failure effects are considered negligible at Browns Ferry Nuclear Plant. Therefore, soil failure effects (such as liquefaction, slope stability and settlement) are considered negligible.

3.0 Prerequisites

The following items have been addressed in order to use the IPEEE analysis for screening purposes and to demonstrate that the IPEEE results can be used for comparison with the Ground Motion Response Spectra (GMRS):

1) Confirmation that commitments made under the IPEEE have been met.
2) Confirmation that all of the modifications and other changes credited in the IPEEE analysis are in place.
3) Confirmation that any identified deficiencies or weaknesses to NUREG-1407 in the Browns Ferry Nuclear Plant Units 1, 2 and 3 IPEEE NRC SER are properly justified to ensure that the IPEEE conclusions remain valid.
4) Confirmation that major plant modifications since the completion of the IPEEE have not degraded/impacted the conclusion reached in the IPEEE.

Response:

Items 1 and 2

- There were no specific commitments identified for Browns Ferry Nuclear Plant as a result of the IPEEE program. There were no modifications identified by the Browns Ferry Nuclear Plant IPEEE report to establish a minimum HCPLF of 0.26g. As noted by the Browns Ferry Nuclear Plant Units 1, 2 and 3 Response to 10 CFR 50.54(f) Request for Information Recommendation 2.3 Seismic (Reference 6.8) Browns Ferry Nuclear Plant performed the IPEEE walkdowns along with Unresolved Safety Issue (USI) A-46. Any adverse condition that was identified during the walkdown was fixed during the USI A-46 plant seismic verification and outlier resolution process.

Item 3

The Browns Ferry Nuclear Units 1, 2 and 3 NRC SER on the seismic portion of the IPEEE submittal identified the following weaknesses in the IPEEE seismic analysis:

- Weakness:
    - The NRC staff identified one weakness for Browns Ferry Nuclear Plant Units 1, 2 and 3 as follows: “seismic IPEEE and noted the success paths do not include any high pressure injection system (reactor core isolation system and high pressure core injection system), and the automatic initial circuitry of the low pressure system.”

The Browns Ferry Nuclear Plant Units 1, 2, and 3 success paths and equipment selected and utilized for the IPEEE programs meet NUREG-1407 (Reference 6.4) and EPRI NP-6041-SL (Reference 6.3) requirements without the high pressure injection system. This is further discussed below in Sections 4.3 and 4.7. Nevertheless, to incorporate NRC recommendations, a confirmatory HCLPF capacity evaluation of the reactor core isolation system is being performed. The results of the HCLPF evaluation for the RCIC system will be included with the activities associated with the Expedited Seismic Evaluation Process (ESEP).

Item 4

A review of major modifications was performed for Browns Ferry Nuclear Plant Units 1, 2 and 3 since the completion of Browns Ferry Nuclear Plant Units 2 and 3 IPEEE developed in 1996. Following this review it was observed that one major modification was generated that potentially impacted the HCLPF capacity of Browns Ferry Nuclear Plant Units 1 and 2. The modification resulted from the addition of a concrete enclosure structure on the roof of the Browns Ferry Nuclear Units 1 and 2 diesel generator structure. As a result of this modification, an updated diesel generator building model was developed. The building responses were only slightly changed. It was observed that the ISRS for the N-S direction was virtually unchanged and the ISRS for the E-W direction was seen to have a decrease in amplitude. Consequently, this modification did not negatively impact the HCLPF capacity of the site.

4.0 Adequacy Demonstration

4.1 Structural Models and Structural Response Analysis

Methodology used:

Structural Models

Major structures for the Browns Ferry Nuclear Plant Units 1, 2 and 3 site considered in the SMA are:

- Reactor building (RB)
- Intake pumping station (IPS)
- Diesel generator buildings (DGBs)

The RB and IPS are both founded on rock and the DGBs are founded on engineered crushed stone that extends to the top of the rock strata at the Browns Ferry Nuclear Plant site.

Between 1986 and 1988, TVA developed updated structural models of all the safety related structures at Browns Ferry Nuclear Plant. These new models were reviewed in detail by the NRC wherein the NRC staff issued their Safety Evaluation Report accepting these new models in NUREG-1232 Volume 3 Supplement 1 (Reference 6.12).

Dynamic models are used to evaluate overall building floor motion due to dynamic loading. The model's physical properties were designed to be used to evaluate overall building floor motion due to dynamic loading. The models are not intended to account for the local amplified response of floors and walls. The models were developed for response spectrum analyses, time history analyses by modal superposition, and the generation of in-structure response spectra.

The dynamic seismic responses were determined using 3-D lumped mass stick models. In general, the mathematical building models have a lumped mass at each major floor elevation in the building. The models consist of massless beam elements and lumped masses at major floor elevations representing floor slabs, walls, equipment and other added weights. The massless beam elements represent the stiffness properties of concrete walls and columns, braced steel column lines and unbraced steel columns. The beam elements, in general, are rigidly linked to each other and to a lumped mass at the center of mass for each major floor elevation. Thus, each floor elevation acts as a rigid body.

The reinforced concrete walls, which respond principally in shear, were condensed to equivalent sets of massless beam elements. Each set of beams mathematically represents the stiffness of one particular functional subset of concrete structural members. Between each floor elevation these beams are geometrically located at the center of shear resistance of the respective structural members which they represent at that floor elevation. These vertical elements are interconnected at the floor levels with massless horizontal rigid beams to affect a total composite elevation response.

The RB analytical model is a multi-stick model that represents the concrete structure and super structure that encompasses the primary containment structure. The RB model also includes the representative mass of the Spent Fuel Pool and its contents. The other sticks of the reactor building model include a stick for the drywell portion of the primary containment, a stick for the biological shield wall, a stick for the reactor vessel support pedestal and a detailed stick representing the reactor pressure vessel, including the reactor pressure vessel internals. Where appropriate, lateral support stiffness springs were included in the reactor building multi stick model to represent various elements such as the star truss, the stabilizer, refueling bellows, and Control Rod Drive (CRD) housing lateral supports. As noted the reactor building is founded on bedrock and thus was modeled as a fixed base structure.

The IPS is founded on rock and thus treated as a fixed base analysis. Additionally, the following should be noted. The effects of soil backfill on the east and west sides of the intake pump station were included in the analysis as soil springs for the E-W direction. The effects of the soil backfill on the north side of the building were included by considering cases with and without soil springs in the N-S direction. The results of these two cases were enveloped to provide response for the N-S motion. Lastly, it should be noted that two seismic analyses were performed considering different reservoir

elevations. The mass of the water enclosed by the structure was included as a lumped mass in the models.
The diesel generator buildings are as noted above, founded on engineered backfill consisting of crushed stone that extends to the top of bedrock at the Browns Ferry Nuclear Plant site.

Consequently, soil amplification studies were performed to derive the input motion at the base of these structures and appropriate soil springs were developed to describe the lateral, vertical and rotation characteristics of the soil column the structure is founded upon.
<u>Structural Response Analysis Method</u>

All Category I structures are analyzed from the three orthogonal component motions (two horizontal and one vertical) of the prescribed earthquake. When the response spectrum analysis is performed, the representative maximum value of a particular response for a N-S or E-W motion (e.g., stress, strain, or displacement) of a given element of a Structure, System, or Component (SSC) is combined with response from the vertical direction.

Compliance with NUREG-1407:

This methodology meets the guidance and requirements of EPRI NP-6041-SL (Reference 6.3) and the enhancements specified in NUREG-1407 (Reference 6.4). Updated structural models developed between 1986 and 1988 using techniques in accordance with NUREG-0800 are consistent with methods outlined in EPRI NP-6041-SL (Section 4 and Appendix E) (Reference 6.3) and the enhancements specified in Section 3.2.5.6 of NUREG-1407 (Reference 6.4).

Adequate for Screening:
The methodology used is in compliance with NUREG-1407 and the IPEEE structural modeling results are adequate for screening purposes.

4.2 In-Structure Demands and ISRS

Methodology used:

The Browns Ferry Nuclear Plant Unit 1,2 and 3 ISRS were developed using the time history method based on 3-D building models for the safety related structures. The ISRS was developed for the horizontal and vertical directions for the resulting structural time history accelerations. The response spectra were peak broadened ± 10 percent for rock-supported structures and ±15 percent for soil-supported structures. A sufficient number of modes of vibration of the structure were included to ensure that a minimum of 90 percent of the building mass participates in the response.

Two site time histories were developed for Browns Ferry Nuclear Plant Units 1, 2 and 3, namely El Centro and artificial earthquake.

> <u>El Centro</u>
>
> The input time history for the horizontal motion is the 1940 El Centro earthquake, N-S component, normalized to 0.1g for OBE and 0.2g for the SSE. The input time history for the vertical direction is 2/3 of the horizontal earthquake record.
>
> <u>Artificial Earthquake</u>
>
> The input time history is an artificial earthquake time history whose spectra envelope the design ground response spectra for all damping values and satisfies the Standard Review Plan 3.7.1 (Reference 6.14) enveloping requirements. The target spectrum for enveloping was the smooth Housner Curve. The vertical input time history is 2/3 of the horizontal time history.

The seismic analysis of primary civil structures is based upon dynamic analysis. The dynamic analyses was performed using input ground motions from the El Centro earthquake to calculate structural response (shear forces , bending moments, axial forces, acceleration and displacements) of the primary civil structures, Reactor Pressure Vessel (RPV) and RPV internals.

An artificial earthquake that closely matches the smooth Housner curve was used to generate amplified response spectra, building accelerations and displacements for subsystem analyses.

Time history modal analyses were used in the seismic analysis of the primary civil structures, whereas response spectrum modal analyses methods were used in the analysis of features, systems, equipment and components. For design purposes, the seismic design combines two earthquake components, one horizontal and one vertical, taken simultaneously. The final design is based upon the more conservative result of the combination of N-S plus vertical response or the E-W plus vertical response.

The RLE in-structure response spectra were developed by scaling the ISRS results from the updated dynamic analysis models of the Browns Ferry Nuclear Plant safety related structures. The scaling factor was determined by the EPRI recommendation of determining the scaling factor by the ratio of spectral amplitudes at dominant structural frequency.

Compliance with NUREG-1407:

This methodology meets the guidance and requirements of EPRI NP-6041-SL (Reference 6.3) and the enhancements specified in NUREG-1407 (Reference 6.4). This method meets the requirements of Section 3.7.1 and 3.7.2 of the Standard Review Plan, NUREG-0800, July 1981 (Reference 6.14), which are consistent with methods outlined in EPRI NP-6041-SL (Section 4 and Appendix E) (Reference 6.3) and the enhancements specified in Section 3.2.5.5 of NUREG-1407 (Reference 6.4).

Adequate for Screening:

The methodology used is in compliance with NUREG-1407 and the IPEEE in-structure demands and ISRS results are adequate for screening purposes.

4.3 Selection of Seismic Equipment List (SEL)/Safe Shutdown Equipment List (SSEL)

Methodology used:

The EPRI methodology (Reference 6.3) was utilized to develop the list of structures, systems and components that would be used for the safety functions required to establish and maintain a safe shutdown condition, including a primary and alternate success path. The following safety functions were satisfied by the IPEEE success paths: reactivity control, reactor coolant pressure control, reactor coolant inventory control, decay heat removal, and containment function.

A list of components was developed for each system with an indication of the component location. The location of equipment was used to ensure that the list of structures was complete for seismic capability screening and analysis.

The types of components considered under the civil/structural review (passive components) were those required to remain intact and provide physical support for mechanical and electrical components.

The passive and active components included in the IPEEE scope are identified in Tables 5-1 and 5-2, and the detailed list of Composite SSEL was included as Appendix B in the Browns Ferry Nuclear Plant Units 1, 2 and 3 IPEEE submittals (References 6.1 and 6.2).

Compliance with NUREG-1407:

This methodology meets the guidance and requirements of EPRI NP-6041-SL (Reference 6.3) and the enhancements specified in NUREG-1407 (Reference 6.4). Section 3.2.5.1 of NUREG-1407 requires a complete set of potential success paths to be identified and the narrowing/elimination of paths to be documented in detail. Section 3.1.2 of the Browns Ferry Nuclear Plant Units 1, 2 and 3 IPEEE (References 6.1 and 6.2) documents in detail the system analysis and the elimination of success paths.

Adequate for Screening:
The methodology used is in compliance with NUREG-1407 and the IPEEE seismic equipment selection results are adequate for screening purposes.

4.4 Screening of Components

Methodology used:

The Seismic Review Team (SRT) screened from further margin review structures and components for which the SRT could document HCLPF capacities at or above the specified Seismic Margin Earthquake (SME) of 0.3g based on their combined experience and judgment and use of earthquake experience data.

The screening guidance given in the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment (GIP) (Reference 6.7) was used. This is the same procedure used for the resolution of USI A-46. This was enhanced for IPEEE in accordance with the EPRI SMA methodology (Reference 6.3) to encompass seismic induced fire and flooding.

Structures and equipment that could not be screened were further evaluated as documented in the IPEEE submittal and supporting calculations and evaluations (References 6.1 and 6.2).

Screening evaluations included spatial interactions, such as assessment of the effects of seismic induced flooding, proximity to other structures or components, etc. (see also walkdown methodology discussion below).

Compliance with NUREG-1407:

The above methodology meets the requirements of NUREG-1407 Section 3.2.5.5 (Reference 6.4) Screening Criteria which states that screening guidance given in the GIP may be used provided review/screening is performed at the appropriate RLE, caveats included in the margins report are observed and use of the generic equipment ruggedness spectrum are observed. NUREG-1407 also requires that spatial interaction evaluations and assessing the effects of flooding as noted in EPRI NP-6041-SL (Reference 6.3) be performed.

Adequate for Screening:

The methodology used is in compliance with NUREG-1407 (Reference 6.4) and the IPEEE screening of component results are adequate for screening purposes.

4.5 Walkdowns

Methodology used:

Although a number of seismic walkdowns were performed as part of the A-46 evaluation, additional walkdowns were performed in support of the IPEEE. The IPEEE scope included passive components and structures, containment isolation and performance and seismic interactions. Walkdowns were documented in walkdown reports and the Screening and Evaluation Walkdown Sheets (SEWS) in accordance with EPRI NP-6041-SL and the Seismic Qualification Utility Group (SQUG).

Walkdowns were conducted by a combination of Seismic Review Team (SRT), who were SQUG trained and certified, and individuals responsible for preparation of the IPE/IPEEE.

The SSEL, based on the success path systems, was used to define the walkdown scope.

Major structures and components were walked down. Emphasis was placed on IPEEE scope not within the A-46 evaluation scope. With the exception of some anchorage, equipment items walked down and accepted for the A-46 evaluation were judged to screened out at 0.3g. Outliers due to anchorage from the A-46 assessment and some new IPEEE scope structures and components were noted as requiring further analysis, HCLPF calculations or modification. Outliers were found acceptable based on further analysis or were modified.

The potential for spatial system interactions was considered during seismic walkdowns. System interaction issues were considered and noted on the SEWS for the IPEEE. The following provides examples of what was considered either previously as part of A-46 walkdowns or as part of the IPEEE:

- Proximity: The proximity of structures to components and components to components was considered during walkdowns. For example, the proximity of valve operators to structures and other components was considered Seismic II over I: examples include consideration of instrument lines and the proximity of block walls to equipment.
- Seismic Spray & Flooding: The possibility of water spray and flooding impact on systems was considered during the walkdown.
- Seismic-induced Fires: The capacity of hydrogen piping and other potential fire hazards was considered as well as proximity to important equipment.
- In addition to the walkdowns performed during the IPEEE review of the plant, Browns Ferry Nuclear Plant implemented a comprehensive II/I seismic interaction program (failure, falling, and impact) as part of the Units 1, 2 and 3 restart efforts. Additionally, a detailed examination of non-safety-related piping systems was

performed to identify any potential breach of the fluid pressure boundary due to its own seismic response or its seismic interaction with other plant features.

Compliance with NUREG-1407:

Walkdowns were conducted and documented in accordance with EPRI NP-6041-SL (Reference 6.3) as required by Section 3.2.5.2 of NUREG-1407 (Reference 6.4).

Adequate for Screening:
The methodology used is in compliance with NUREG-1407 (Reference 6.4) and the IPEEE walkdown results are adequate for screening purposes.

4.6 Fragility Evaluations

Methodology used:

Analyses were performed for all structures and components that could not be screened out to a HCLPF capacity review level earthquake of 0.3g. The seismic capacities of structures and components were calculated in accordance with the guidance contained in the GIP (Reference 6.7) and EPRI NP-6041-SL (Reference 6.3) and are documented in References 6.1 and 6.2.

The ground response spectra for the Browns Ferry Nuclear Units 1, 2 and 3 IPEEE is the NUREG/CR-0098 (Reference 6.11) rock median spectral shape according to NUREG-1407 for seismic margin evaluations. The floor response spectra were developed based on Reference 6.3.

All equipment and structures in the SMA success path have a reported HCLPF capacity of 0.3g PGA or greater except for two 4KV/480V transformers located in the Unit 2 diesel generator rooms at elevation 583 which had reported HCLPF capacity of 0.26g. Equipment and structures evaluated for Units 1 and 3 had a reported HCLPF of at least 0.3 g. It should be noted that one of the transformers has been replaced with a more robust transformer and the second transformer is scheduled to be replaced in the fall of 2014 (Reference 6.8).

Compliance with NUREG-1407:

Browns Ferry Nuclear Units 1, 2 and 3 calculated HCLPF capacities for all outlier components in accordance with the guidance of EPRI NP-6041-SL (Reference 6.3) and NUREG-1407 Section 3.2.5.7 (Reference 6.4). Components that did not meet the 0.3g RLE screening criteria were modified except for one of the two 4KV/480V transformers located in the Unit 2 diesel generator rooms at elevation 583 as identified above that is scheduled to be replaced in the fall of 2014.

Adequate for Screening:

The methodology used is in compliance with NUREG-1407 (Reference 6.4) and the IPEEE fragility evaluation results are adequate for screening purposes.

4.7 System Modeling

Methodology used:

Functional success paths were developed with the aid of the Individual Plant Examinations (IPE) Probabilistic Risk Assessment (PRA) event tree models to identify systems needed to mitigate the consequences of an earthquake. The functional success diagram shows the front line and the associated support systems that can be used for the safety functions required to establish and maintain a long term safe shutdown condition (i.e., reactivity control, pressure and inventory control and decay heat removal).

All potential success paths were evaluated and eliminated based on not meeting the 72-hour mission time, the inability to meet the review level earthquake, the inability to cope with a small break loss of coolant accident, etc. The systems considered and the reasons for being eliminated are documented in the Browns Ferry Nuclear Plants Units 1, 2 and 3 IPEEE submittals (References 6.1 and 6.2).

The evaluations of non-seismic failures and human actions were considered, although not explicitly addressed in the IPEEE evaluation of seismic risk. The systems and components in the success path with the highest non-seismic unreliability were identified and the impact on risk was evaluated and documented in the IPEEE.

The identification of success paths and components was based on, in all cases, operator actions required to achieve Safe Shutdown as normally credited in the operator training programs. There were no new operator actions required to support the primary and alternate success paths identified and evaluated in the IPEEE.

Compliance with NUREG-1407:

NUREG-1407, Section 3.2.5.1 (Reference 6.4) states that for IPEEE purposes, it is desirable that to the maximum extent possible, the alternate path involves operational sequences, systems, piping runs and components different from those used in the preferred path. As indicated above and documented in the IPEEE, this requirement was met based on the design of Browns Ferry Nuclear Units 1, 2 and 3.

The treatment of non-seismic failures and human actions in the Browns Ferry Nuclear Plant Units 1, 2 and 3 IPEEE meets the requirements of Section 3.2.5.8 of NUREG-1407 (Reference 6.4).

Adequate for Screening:

The methodology used is in compliance with NUREG-1407 (Reference 6.4) and the IPEEE system modeling results are adequate for screening purposes.

4.8 Containment Performance

Methodology used:

Containment performance was evaluated from a structural, isolation and bypass perspective. The containment structure [drywell and suppression chamber (torus)] was evaluated using the methodology and guidance of EPRI NP-6041-SL (Reference 6.3). The structure was found to be seismically rugged.

For containment isolation and bypass, the containment penetration screening analysis in the IPE was utilized. It was determined that isolation valves are seismically rugged and/or the systems outside containment are seismically rugged and designed for high pressure.

No containment vulnerabilities were found.

Compliance with NUREG-1407:

The review of containment meets the requirements of Section 3.2.6 of NUREG-1407 (Reference 6.4) to evaluate the containment integrity, isolation, bypass and suppression functions to identify vulnerabilities that involve early failure of the containment functions.

Adequate for Screening:

The methodology used is in compliance with NUREG-1407 (Reference 6.4) and the IPEEE containment performance results are adequate for screening purposes.

4.9 Peer Review

Methodology used:

An in-house independent review team was established outside the IPEEE team. The independent review team consisted of a cross disciplinary review by structural, electrical and mechanical engineering groups. All reviewer comments were addressed by the IPEEE team.

In addition to the in-house review team, external consultants were called upon to participate in the Peer Review efforts.

The Browns Ferry Nuclear Plant Unit 2 and 3 Peer Review was performed by Mr. Greg Hardy of EQE, and Mr. Richard Cutsinger of TVA. Mr. Hardy is recognized seismic margin expert who was a consultant to EPRI in the development of the GIP for resolution of USI a-46. Mr. Hardy was also involved with the performance of many A-46 and SMA evaluations for nuclear facilities. Mr. Cutsinger was the TVA Chief Civil Engineer who was the Final Technical Authority for all civil activities associated with the TVA nuclear fleet. Mr. Cutsinger was also a member of the Seismic Qualification Utility Group (SQUG) Steering Committee. The Browns Ferry Nuclear Plant Unit 1 Peer Review was performed by Dr. James J. Johnson. Dr. Johnson is a recognized seismic expert, who authored several papers on SSI analysis of nuclear facilities as well as development of SME ISRS at several nuclear facilities. Dr. Johnson was also very involved with the development of EPRI NP6041-SL R1.

The Unit 2 and 3 Peer Review report identified an issue related to methods used to scale the 0.3g seismic margin spectra. The scaling methods were modified to be consistent with the EPRI SMA methodology (Reference 6.3).

The Browns Ferry Nuclear Plant Unit 1 Peer Review report suggested that the definition of the safe shutdown paths and the safe shutdown equipment lists for the three units should be maintained to the extent possible. Also, shared systems for the multiple units have the capacity to meet the demand of the multiple units when simultaneously subjected to the event of interest.

Compliance with NUREG-1407:

The above review process, using a combination of IPEEE Team Members, an independent In-house Review Team and an external consultant for seismic review, meet the requirements of Section 7 of NUREG-1407 (Reference 6.4) for peer review.

Adequate for Screening:

The methodology used is in compliance with NUREG-1407 (Reference 6.4) and the IPEEE peer review results are adequate for screening purposes.

5.0 Conclusion

The Browns Ferry Nuclear Plant Unit 1,2, and 3 IPEEE was a focused scope margin submittal and requires the performance of a detailed review of relay chatter and full evaluation of soil failures to be considered as a full-scope assessment. A soil failure evaluation has been completed as noted in Section 2.2 with satisfactory results. A relay evaluation consistent with a full scope IPEEE, as described in NUREG-1407 (Reference 6.4), will be performed on the schedule provided in NEI letter to NRC dated October 3, 2013 (Reference 6.9).

Based on the IPEEE adequacy review performed consistent with the guidance contained in Reference 6.18 and documented herein, with the exception of the completion of the detailed relay chatter review, the Browns Ferry Nuclear Plant Units 1, 2 and 3 IPEEE results are considered adequate for screening and the risk insights gained from the IPEEE remain valid under the current plant configuration. Based on the results of this IPEEE adequacy review it can be concluded that a minimum HCLPF capacity of 0.26g can be utilized for IHS screening purposes for Browns Ferry Nuclear Plant.

6.0 References

6.1 TVA Letter from P. Salas to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Generic Letter (GL) 87-02, Supplement 1, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46 and GL 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Submittal of Seismic Evaluation Reports (TAC Nos. M69431, M69432, M83596 and M83587)," dated June 28, 1996.

6.2 TVA Letter from T. E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 Response to NRC Generic Letter (GL) 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Submittal of Browns Ferry Nuclear Plant Unit 1 Seismic and Internal Fires IPEEE Reports" dated January 14, 2005.

6.3 EPRI NP-6041-SL Revision 1, A Methodology of Assessment of Nuclear Power Plant Seismic Margin (Revision 1), dated August 1991

6.4 NUREG-1407 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Sever Accident Vulnerabilities" - Final Report, June 1991

6.5 Letter to J. A. Scalice from NRC, "Browns Ferry Units 1, 2 and 3, Individual Plant Examination of External Events (IPEEE) and Related Generic Safety Issues, Issuance of Staff Evaluation (TAC Nos. M83595, M83596, M83679)," June 22, 2000.

6.6 Letter to William R. Campbell from NRC, "Browns Ferry Nuclear Plant, Unit 1 - Closeout of Generic Letter 88-20, Supplement 4, Concerning Individual Plant Examination of External Events for Severe Accident Vulnerabilities (TAC No. MC5729)," June 28, 2007.

6.7 "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 2, February 14, 1992.

6.8 Letter from J. W. Shea to NRC, "Tennessee Valley Authority (TVA) - Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding the Browns Ferry Nuclear Plant Seismic Walkdown Results of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," November 27, 2012.

6.9 NEI Letter from Kimberly A. Keithline to David L. Skeen, NRC, "Relay Chatter Reviews for Seismic Hazard Screening," October 3, 2013.

6.10 Letter from J. Shea to NRC, “Tennessee Valley Authority - Supplemental Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding the Browns Ferry Nuclear Plant, Unit 2 Seismic Walkdown Results of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” June 28, 2013.

6.11 NUREG/CR-0098, Development of Criteria for Seismic Review of Selected Nuclear Power Plants, dated May 1978

6.12 NUREG-1232, Volume 3, Supplement 1, “Safety Evaluation Report on Tennessee Valley Authority Nuclear Performance Plan,” July 1987.

6.13 “Liquefaction Susceptability Criteria for Silts and Clays,” by Ross Boulanger and I. M. Idriss, Journal of Getechnical and Geoenvironmental Engineering, ASCE, November, 2006.

6.14 NUREG-0800, “Standard Review Plan,” Section 3.7.2, July 1981.

6.15 U.S. Nuclear Regulatory Commission, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR 50.54(f) (Generic Letter No. 88-20, Supplement 4),” June 28, 1991.

6.16 10 CFR Part 50. Title 10, Code of Federal Regulations, Part 50, “Domestic Licensing of Production and Utilization Facilities,” U.S. Nuclear Regulatory Commission, Washington DC.

6.17 10 CFR Part 100. Title 10, Code of Federal Regulations, Part 100, “Reactor Site Criteria,” U.S. Nuclear Regulatory Commission, Washington DC.

6.18 Electric Power Research Institute Report 1025287, “Seismic Evaluation Guidance Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” February 2013.

6.19 NEI Letter to NRC, “Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations,” April 9, 2013.

6.20 NRC Letter, Eric J. Leeds to Joseph E. Pollock, NEI “Electric Power Research Institute Final Draft Report XXXXXX, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, As an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluation,” May 7, 2013.

6.21 “Browns Ferry Nuclear Power Station – Final Safety Analysis Report,” Amendment 25.3.

6.22 “Seismic Data Retrieval Information for EPRI Near-Term Task Force Recommendation 2.1 TVA Browns Ferry Nuclear Plant Athens, Alabama,” AMEC Project 3043121013 report transmitted by letter from K. Campbell to J. Best, June 26, 2013.

ENCLOSURE 3

SEISMIC HAZARD AND SCREENING REPORT FOR TENNESSEE VALLEY AUTHORITY'S SEQUOYAH NUCLEAR PLANT

Seismic Hazard and Screening Report for Sequoyah Nuclear Plant

Table of Contents

1.0 Introduction

Following the accident at the Fukushima Daiichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near-Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter (U.S. NRC, 2012) that requests information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter (U.S. NRC, 2012) requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements. Depending on the comparison between the reevaluated seismic hazard and the current design basis, the result is either no further risk evaluation or the performance of a seismic risk assessment. Risk assessment approaches acceptable to the staff include a seismic probabilistic risk assessment (SPRA), or a seismic margin assessment (SMA). Based upon this information, the NRC staff will determine whether additional regulatory actions are necessary.

This report provides the information requested in items (1) through (7) of the "Requested Information" section and Attachment 1 of the 50.54(f) letter (U.S. NRC, 2012) pertaining to NTTF Recommendation 2.1 for the Sequoyah Nuclear Plant, located in Hamilton County, Tennessee. In providing this information, the Tennessee Valley Authority followed the guidance provided in the *Seismic Evaluation Guidance: Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic* (EPRI, 2013a). The Augmented Approach, *Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic* (EPRI, 2013c), has been developed as the process for evaluating critical plant equipment as an interim action to demonstrate additional plant safety margin prior to performing the complete plant seismic risk evaluations.

The original geologic and seismic siting investigations for Sequoyah Nuclear Plant was performed in accordance with Appendix A to 10 CFR Part 100 and meet General Design Criterion 2 in Appendix A to 10 CFR Part 50. The Safe Shutdown Earthquake (SSE) Ground Motion was developed in accordance with Appendix A to 10 CFR Part 100 and used for the design of seismic Category I Systems, Structures, and Components (SSCs).

In response to the 50.54(f) letter (U.S. NRC, 2012) and following the guidance provided in the SPID (EPRI, 2013a), a seismic hazard reevaluation for Sequoyah Nuclear Plant was performed. For screening purposes, a Ground Motion Response Spectrum (GMRS) was developed. Based on the results of the screening evaluation, Sequoyah Nuclear Plant screens-in for a risk evaluation, a Spent Fuel Pool evaluation, and a High Frequency Confirmation.

2.0 Seismic Hazard Reevaluation

Sequoyah Nuclear Plant site is located approximately 7.5 miles northeast of the nearest city limit of Chattanooga, Tennessee, on a peninsula on the western shore of Chickamauga Lake at Tennessee River mile marker 484.5 (TVA, Amendment 24, Section 2.1.1). The Sequoyah Nuclear Plant site is located in the Appalachian Valley subregion of the Valley and Ridge Province of the Appalachian Highlands. Physiographically, this subregion is characterized by long narrow ridges and somewhat broader intervening valleys having a northeast-southwest trend. The ridges are roughly parallel and fairly evenly topped. They are developed in areas underlain by resistant sandstones and the more siliceous limestones and dolomites. The valleys have been excavated in the areas underlain by easily weathered shales and the more soluble limestone formations. (TVA, Amendment 24, Section 2.5.1.2)

The evaluation of the earthquake hazard at the Sequoyah Nuclear Plant site involves a consideration of the known seismic history of a large surrounding area. By plotting the epicenters of hundreds of earthquake shocks, the areas of continuing seismic activity become apparent. The more active areas are described in the following summary. (TVA, Amendment 24, Section 2.5.2.3)

a. Mississippi Valley, especially the New Madrid region of Arkansas, Kentucky, Missouri, and Tennessee. A few great earthquakes and thousands of light to moderately strong shocks have been centered in the Mississippi Valley. Light to moderate shocks are still occurring at an average frequency of a few per year. The New Madrid region is more than 250 miles northwest of the Sequoyah Nuclear Plant site.
b. The Lower Wabash Valley of Illinois and Indiana. This area has been the center of several moderately strong earthquakes, some of which were felt as far south as Nashville, Tennessee. It is about 260 miles northwest of the Sequoyah Nuclear Plant site.
c. Charleston area, South Carolina. One of the country's greatest earthquakes was centered in the Charleston area. Earlier, many light to moderate shocks had been centered in the area long before the great earthquake, and the activity has continued to the present time. Charleston is more than 300 miles east of the Sequoyah Nuclear Plant site.
d. The Appalachian Mountains of eastern Tennessee and western North Carolina. The mountain belt of eastern Tennessee and western North Carolina is a region of continuing minor activity. Light to moderate shocks occur at an average frequency of one or two per year. The activity is not uniform, as periods of several shocks per year are followed by longer periods of no perceptible shocks. This region is centered more than 50 miles to the east of the Sequoyah Nuclear Plant site.

In addition to these areas, shocks of light to moderate intensity have occurred at numerous other localities in the southeastern states at various distances from the Sequoyah Nuclear Plant site. At many of these localities, only a few light to moderate shocks from widely scattered epicenters are known. A few such shocks have occurred to the north and east of Huntsville, Alabama. Numerous light shocks have occurred in Knoxville and its environs. (TVA, Amendment 24, Section 2.5.2.3)

The maximum historic quake reported in this Province was assigned an intensity of VIII on the Modified Mercalli Intensity Scale of 1931 although there is reason to believe it should have been rated as intensity VII on the Modified Mercalli Intensity Scale of 1931. It occurred in Giles County, Virginia, in 1897. Although this earthquake occurred 285 miles northeast of the site, this intensity is assumed to occur at the site for the purpose of defining the SSE. The maximum acceleration for an intensity of this level is estimated to be 0.14g. The Sequoyah Nuclear Plant is designed so that all SSCs important to safety will remain functional when subjected to an SSE having maximum horizontal acceleration of 0.18g and maximum vertical ground acceleration of 0.12g. (TVA, Amendment 24, Section 2.5.2.4)

2.1 Regional and Local Geology

The Sequoyah Nuclear Plant site is located in the Appalachian Valley subregion of the Valley and Ridge Province of the Appalachian Highlands. (TVA, Amendment 24, Section 2.5.1.2) The controlling features of the geologic structure at the Sequoyah Nuclear Plant site are the Kingston Thrust fault and a major overturned anticline which resulted from the movement along the fault. This fault lies about a mile northwest of the plant site and can be traced for 75 miles northeastward and 70 miles southwestward. The fault dips to the southeast, under the plant site, and along it steeply dipping beds of the Knox dolomite have been thrust over gently dipping strata of the Chickamauga limestone. The distance from the plant site, about one mile, and the dip of the fault, 30 degrees or more, will carry the plane of the fault at least 2000 feet below the surface at the plant site. (TVA, Amendment 24, Section 2.5.1.5)

The major overturned anticline results in the Conasauga formation at the plant site resting upon the underlying Knox dolomite which normally overlies it. As a result of the ancient structural movement of the fault and major fold, the Conasauga formation at the plant site is highly folded, complexly contorted, and cut by many very small subsidiary faults and shears. The general strike of these beds are N 30 degrees E and the overall dip is to the southeast, but the many small tightly folded, steeply pitching anticlines and synclines result in many local variations to the normal trend. (TVA, Amendment 24, Section 2.5.1.5)

The Kingston fault is only one of the several lengthy thrust faults which characterize the geologic structure of the Appalachian Valley, a part of the "Valley and Ridge" physiographic province. A study of any one of these faults involves a consideration of the major structural features of the Valley as a whole. Structurally, the Appalachian Valley in eastern Tennessee is characterized very largely by a series of overlapping linear fault blocks of northeast-southwest strike and southeast dips. (TVA, Amendment 24, Section 2.5.1.5)

The only undeformed materials occurring in the Valley as mappable units are the unconsolidated materials: alluvial deposits, including the high level terrace deposits as well as the recent floodplain alluvium, and the residuum that nearly everywhere mantles bedrock. (TVA, Amendment 24, Section 2.5.1.5)
Most major Category I structures are founded on bedrock and no subsidence is to be expected. In most instances the weight of rock removed in foundation excavation equals or exceeds the weight imposed by the structure. Sufficient exploratory drilling has been done to assure there are no karstic solution zones underlying the plant that would allow collapse. Any small solution areas below foundation grade have been grouted in the routine course of construction. (TVA, Amendment 24, Section 2.5.4.1)

2.2 Probabilistic Seismic Hazard Analysis

2.2.1 Probabilistic Seismic Hazard Analysis Results

In accordance with the 50.54(f) letter (U.S. NRC, 2012) and following the guidance in the SPID (EPRI, 2013a), a probabilistic seismic hazard analysis (PSHA) was completed using the recently developed Central and Eastern United States Seismic Source Characterization (CEUS-SSC) for Nuclear Facilities (CEUS-SSC, 2012) together with the updated Electric Power Research Institute (EPRI) Ground-Motion Model (GMM) for the Central and Eastern United States (CEUS) (EPRI, 2013b). For the PSHA, a lower-bound moment magnitude of 5.0 was used, as specified in the 50.54(f) letter (U.S. NRC, 2012). (EPRI, 2014)

For the PSHA, the CEUS-SSC background seismic sources out to a distance of 400 miles (640 km) around Sequoyah Nuclear Plant were included. This distance exceeds the 200 mile (320 km) recommendation contained in Reg. Guide 1.208 (U.S. NRC, 2007) and was chosen for completeness. Background sources included in this site analysis were the following (EPRI, 2014):

1. Extended Continental Crust—Atlantic Margin (ECC_AM)
2. Extended Continental Crust—Gulf Coast (ECC_GC)
3. Illinois Basin Extended Basement (IBEB)
4. Mesozoic and younger extended prior – narrow (MESE-N)
5. Mesozoic and younger extended prior – wide (MESE-W)
6. Midcontinent-Craton alternative A (MIDC_A)
7. Midcontinent-Craton alternative B (MIDC_B)
8. Midcontinent-Craton alternative C (MIDC_C)
9. Midcontinent-Craton alternative D (MIDC_D)
10. Non-Mesozoic and younger extended prior – narrow (NMESE-N)
11. Non-Mesozoic and younger extended prior – wide (NMESE-W)
12. Paleozoic Extended Crust narrow (PEZ_N)
13. Paleozoic Extended Crust wide (PEZ_W)
14. Reelfoot Rift (RR)
15. Reelfoot Rift including the Rough Creek Graben (RR-RCG)

16. Study region (STUDY_R)

For sources of large magnitude earthquakes (designated Repeated Large Magnitude Earthquake (RLME) sources), in NUREG-2115 (CEUS-SSC, 2012) modeled for the CEUS-SSC, the following sources lie within 1,000 km of the site and were included in the analysis (EPRI, 2014):

1. Charleston
2. Commerce
3. Eastern Rift Margin Fault northern segment (ERM-N)
4. Eastern Rift Margin Fault southern segment (ERM-S)
5. Marianna
6. New Madrid Fault System (NMFS)
7. Wabash Valley

For each of the above background and RLME sources, the mid-continent version of the updated CEUS EPRI GMM was used. (EPRI, 2014)

2.2.2 Base Rock Seismic Hazard Curves

Consistent with the SPID (EPRI, 2013a), base rock seismic hazard curves are not provided as the site amplification approach referred to as Method 3 has been used. Seismic hazard curves are shown below in Figure 2.3.7-1. (EPRI, 2014)

2.3 Site Response Evaluation

Following the guidance contained in Seismic Enclosure 1 of 50.54(f) Request for Information (U.S. NRC, 2012) and in the SPID (EPRI, 2013a) for nuclear power plant sites that are not founded on hard rock (defined as 2.83 km/sec), a site response analysis was performed for Sequoyah Nuclear Plant. (EPRI, 2014)

2.3.1 Description of Subsurface Material

Sequoyah Nuclear Plant is located on the western shore of Chickamauga Lake in Hamilton County, Tennessee. The site is located in the Tennessee section of the Appalachian Valley subregion of the Valley and Ridge Province of the Appalachian Highlands. The subregion is characterized by long narrow ridges and somewhat broader intervening valleys with a northeast to southwest trend (AMEC, 2013). (EPRI, 2014)

The plant is founded on complexly folded, interbedded limestone and shale bedrock of the Conasauga Formation of Middle Cambrian age. The information used to create the site geologic profile at Sequoyah Nuclear Plant is shown in Tables 2.3.1-1 and 2-3.1-2. This profile was developed using information documented in AMEC (2013). As indicated in AMEC (2013) the SSE Control Point is at a depth of 64 ft, and the profile was modeled up to this location. (EPRI, 2014)

Tables 2.3.1-1 and 2.3.1-2 show the recommended geotechnical properties for the site. (EPRI, 2014)

Table 2.3.1-1. Summary of Site Geotechnical Profile for Sequoyah Nuclear Plant (AMEC, 2013). (EPRI, 2014)

Depth (ft)	Soil/Rock Description	Density (lb/ft^3)	Measured V_s* (ft/s)	V_s for Analyses (ft/s)	G_{max} (lb/ft^2)	G/G_{max} vs. Shear Strain	Damping Ratio vs. Shear Strain
0	Ground Surface Elev. 705	–	–	–	–	–	–
0 – 38	Residual Clays and Silts**	115	442 – 3,050 Average 1,180	1,200	3,700,000	Use Watts Bar FSAR Figure 2.5-233E	Use Watts Bar FSAR Figure 2.5-233F
38 – 64	Limestone with interbedded Shale	170	4,873 – 9,697 Average 6,723	6,700 (±1,000)	237,000,000	1	No Change
64	**Deepest Structure Foundation Control Point – SSE GMRS**	–	–	–	–	–	–
64 – 103	Limestone with interbedded Shale	170	4,873 – 9,697 Average 6,723	6,700 (±1,000)	237,000,000	1	No Change

Notes: *The range of shear-wave velocities measured in various geophysical tests performed at the site.
**Replaced with engineered backfill for safety related structures.

Table 2.3.1-2. Summary of Geologic Profile for Sequoyah Nuclear Plant Extended to Basement (AMEC, 2013). (EPRI, 2014)

Depth (ft)	Soil/Rock Description*	Rock Formation	Best Estimate V_s (ft/s)**	Lower Range V_s (ft/s)***	Upper Range V_s (ft/s)***
0 – 1,500	Shale, light-green to brown; limestone, medium-gray, dolomitic, coarse-grained, oolitic, and commonly conglomeratic; lower part consists of shale and siltstone.	Єc – Conasauga Group, Undivided	6,000	4,800	7,500
-	-	Kingston Fault			
1,500 – 1,650	Upper part consists of greenish- gray and grayish-red calcisiltite and claystone. Lower part is a light-gray, thick-bedded calcilutite. A basal conglomerate, occurring locally, is slight greenish- to reddish- gray dolosiltite, with thin- to medium-bedded, light greenish- gray calcilutite and calcisiltite, or lenses of shale and sandstone.	Ops – Pond Spring Formation	9,500	6,050	9,285
1,650 – 4,800	Dolomite and minor limestone, very siliceous, light- to dark-gray, fine- to coarse-grained, thin- to thick- bedded, weathers to cherty rubble. Thickness about 2,600 ft.	OЄk – Knox Group, Undifferentiated	7,000	4,460	9,285
4,800 – 6,250	Shale, light-green to brown; limestone, medium-gray, dolomitic, coarse-grained, oolitic, and commonly conglomeratic; lower part consists of shale and siltstone.	Єc – Conasauga Group, Undivided	7,000	4,460	9,285
6,250 – 7,580	Consists of sandstone, siltstone, and shale. Formation not exposed; shown in structure section only.	Єr – Rome Formation	10,000	6,370	9,285

Table 2.3.1-2. Summary of Geologic Profile for Sequoyah Nuclear Plant Extended to Basement (AMEC, 2013), Continued. (EPRI, 2014)

Depth (ft)	Soil/Rock Description*	Rock Formation	Best Estimate V_s (ft/s)**	Lower Range V_s (ft/s)***	Upper Range V_s (ft/s)***
-	-	Chattanooga Fault			
7,580 – 9700	Dolomite and minor limestone, very siliceous, light- to dark-gray, fine- to coarse-grained, thin- to thick- bedded, weathers to cherty rubble. Thickness about 2,600 ft.	OЄk – Knox Group, Undifferentiated	7,000	4,460	9,285
9,700 – 11,150	Shale, light-green to brown; limestone, medium-gray, dolomitic, coarse-grained, oolitic, and commonly conglomeratic; lower part consists of shale and siltstone.	Єc – Conasauga Group, Undivided	7,000	4,460	9,285
11,150 – 11,900	Consists of sandstone, siltstone, and shale. Formation not exposed; shown in structure section only.	Єr – Rome Formation	10,000	6,370	9,285
-	-	Sequatchie Valley Fault			
11,900 – 12,350	Consists of sandstone, siltstone, and shale. Formation not exposed; shown in structure section only.	Єr – Rome Formation	10,000	6,370	9,285
>12,350	-	Basement	12,000	7,640	9,285

*Note: Rock Descriptions obtained from Drahovzal and Neathery (1969) and Lemiszki, et al. (2008).

**Note: These values were based on Spectral-Analysis-of-Surface-Waves (SASW) testing by Dr. Ken Stokoe at the Watts Bar nuclear plant site, which consists of similar rock formation to base these values upon. Ivan Wong from URS assisted Dr. Stokoe and AMEC in developing a lognormal average for the best estimate.

***Note: The lower and upper ranges were based on the best estimate, with the upper range constrained not to exceed 9,285 ft/s. For depths of 0 – 1,500 ft, these values were calculated using a certainty of 1.25. For depths of 1500 ft to basement, these values were calculated using a certainty of 1.57.

2.3.2 Development of Base Case Profiles and Nonlinear Material Properties

Tables 2.3.1-1 and 2.3.1-2 show the recommended shear-wave velocities and unit weights versus depth and stratigraphy for the profile. Based on Tables 2.3.1-1 and 2.3.1-2, and the location of the SSE at a depth of 64 ft (19.5 m) (AMEC, 2013), the profile consists of 6,186 ft (1,885 m) of firm rock overlying hard crystalline basement rock. (EPRI, 2014)

Shear-wave velocities for the profile were made in some borings at the site to a depth of 103 ft (31 m). Shear-wave velocities ranged from 4,873 ft/s (1,485 m/s) to 9,697 ft/sec (2,955 m/s) (Table 2.3.1-1). There were no trends in these velocity measurements to suggest an increase in shear-wave velocity with depth in the Conasauga Shale (AMEC, 2013). More recent SASW measurements at Watts Bar Nuclear Plant on the same formations are shown in Table 2.3.1-2 (AMEC, 2013). (EPRI, 2014)

Based on the specified range in measured shear-wave velocities in the top 38 ft (12 m) beneath the SSE (Table 2.3.1-1), a scale factor of 1.57 was adopted to reflect upper and lower range base-cases. The scale factor of 1.57 reflects a $\sigma_{\mu \ln}$ of about 0.35 based on the SPID (EPRI, 2013a) 10^{th} and 90^{th} fractiles which implies a 1.28 scale factor on σ_{μ}. (EPRI, 2014)

Using the best estimate or mean base-case profile (P1), the depth-independent scale factor of 1.57 was applied to develop lower and upper range base-cases profiles P2 and P3 respectively with the stiffest profile (P3) reaching hard reference rock velocities at the surface. Base-case profiles P1 and P2 have a mean depth below the SSE of 6,186 ft (1,885 m) to hard reference rock, taken at the Rome Formation (Table 2.3.1-2) and randomized ±1,885 ft (±566 m). The base-case profiles (P1, P2, and P3) are shown in Figure 2.3.2-1 and listed in Table 2.3.2-1. The depth randomization reflects ±30% of the depth to provide a realistic broadening of the fundamental resonance rather than reflect actual random variations to basement shear-wave velocities across a footprint. (EPRI, 2014)

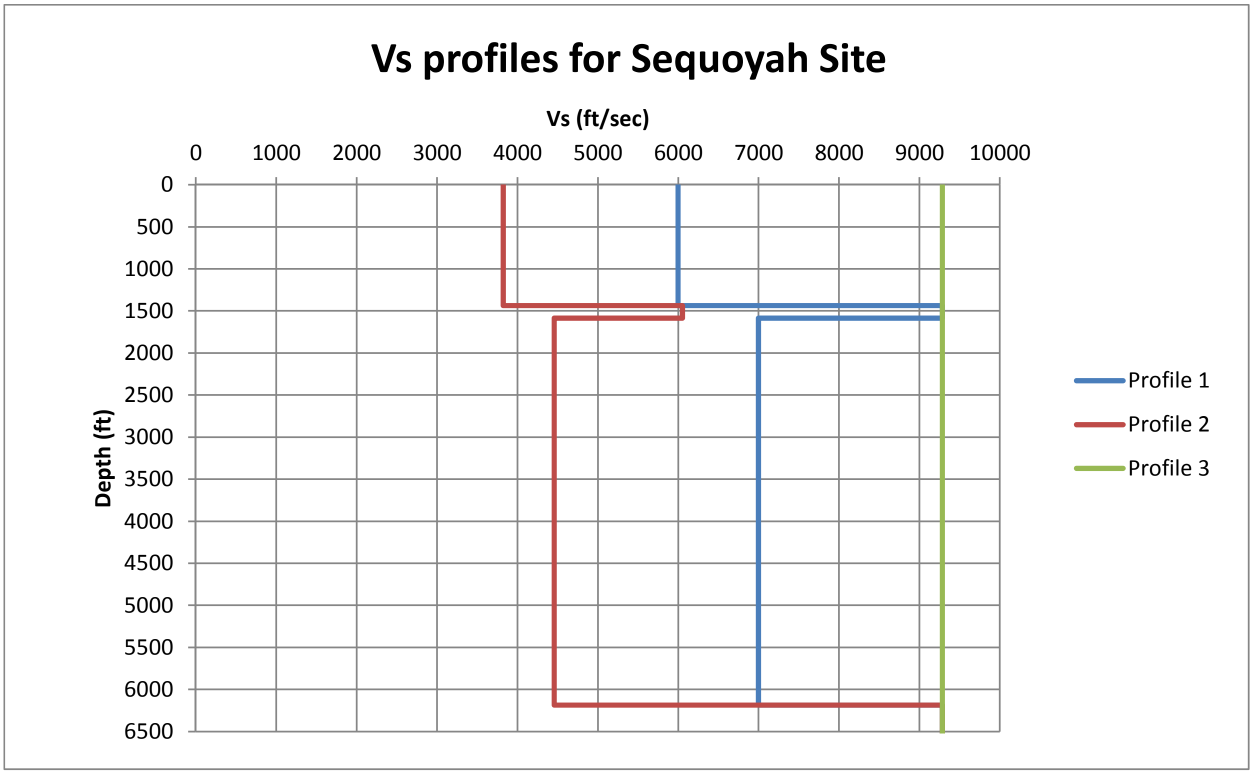


Figure 2.3.2-1. Shear-wave velocity profiles for Sequoyah Nuclear Plant. (EPRI, 2014)

Table 2.3.2-1. Layer thicknesses, depths, and shear-wave velocities (V_s) for 3 profiles, Sequoyah Nuclear Plant. (EPRI, 2014)

Profile 1			Profile 2			Profile 3		
thickness (ft)	depth (ft)	V_s (ft/s)	thickness (ft)	depth (ft)	V_s (ft/s)	thickness (ft)	depth (ft)	V_s (ft/s)
	0	6000		0	3821		0	9285
6.7	6.7	6000	6.7	6.7	3821	6.7	6.7	9285
6.7	13.3	6000	6.7	13.3	3821	6.7	13.3	9285
6.7	20.0	6000	6.7	20.0	3821	6.7	20.0	9285
10.0	30.0	6000	10.0	30.0	3821	10.0	30.0	9285
10.0	40.0	6000	10.0	40.0	3821	10.0	40.0	9285
10.0	50.0	6000	10.0	50.0	3821	10.0	50.0	9285
10.0	60.0	6000	10.0	60.0	3821	10.0	60.0	9285
10.0	70.0	6000	10.0	70.0	3821	10.0	70.0	9285
10.0	80.0	6000	10.0	80.0	3821	10.0	80.0	9285
10.0	90.0	6000	10.0	90.0	3821	10.0	90.0	9285
10.0	100.0	6000	10.0	100.0	3821	10.0	100.0	9285
10.0	110.0	6000	10.0	110.0	3821	10.0	110.0	9285
10.0	120.0	6000	10.0	120.0	3821	10.0	120.0	9285
13.0	133.0	6000	13.0	133.0	3821	13.0	133.0	9285
13.0	146.0	6000	13.0	146.0	3821	13.0	146.0	9285
13.0	159.0	6000	13.0	159.0	3821	13.0	159.0	9285

Table 2.3.2-1. Layer thicknesses, depths, and shear-wave velocities (V_s) for 3 profiles, Sequoyah Nuclear Plant, Continued. (EPRI, 2014)

Profile 1			Profile 2			Profile 3		
thickness (ft)	depth (ft)	V_s (ft/s)	thickness (ft)	depth (ft)	V_s (ft/s)	thickness (ft)	depth (ft)	V_s (ft/s)
13.0	172.0	6000	13.0	172.0	3821	13.0	172.0	9285
13.0	185.0	6000	13.0	185.0	3821	13.0	185.0	9285
13.0	198.0	6000	13.0	198.0	3821	13.0	198.0	9285
13.0	211.0	6000	13.0	211.0	3821	13.0	211.0	9285
13.0	224.0	6000	13.0	224.0	3821	13.0	224.0	9285
13.0	237.0	6000	13.0	237.0	3821	13.0	237.0	9285
13.0	250.0	6000	13.0	250.0	3821	13.0	250.0	9285
25.0	275.0	6000	25.0	275.0	3821	25.0	275.0	9285
25.0	300.0	6000	25.0	300.0	3821	25.0	300.0	9285
25.0	325.0	6000	25.0	325.0	3821	25.0	325.0	9285
25.0	350.0	6000	25.0	350.0	3821	25.0	350.0	9285
25.0	375.0	6000	25.0	375.0	3821	25.0	375.0	9285
25.0	400.0	6000	25.0	400.0	3821	25.0	400.0	9285
25.0	425.0	6000	25.0	425.0	3821	25.0	425.0	9285
25.0	450.0	6000	25.0	450.0	3821	25.0	450.0	9285
25.0	475.0	6000	25.0	475.0	3821	25.0	475.0	9285
25.0	500.0	6000	25.0	500.0	3821	25.0	500.0	9285
36.0	536.0	6000	36.0	536.0	3821	36.0	536.0	9285
225.0	761.0	6000	225.0	761.0	3821	225.0	761.0	9285
225.0	985.9	6000	225.0	985.9	3821	225.0	985.9	9285
225.0	1210.9	6000	225.0	1210.9	3821	225.0	1210.9	9285
225.0	1435.9	6000	225.0	1435.9	3821	225.0	1435.9	9285
75.0	1510.9	9285	75.0	1510.9	6051	75.0	1510.9	9285
75.0	1585.9	9285	75.0	1585.9	6051	75.0	1585.9	9285
225.0	1810.9	7000	225.0	1810.9	4458	225.0	1810.9	9285
225.0	2035.9	7000	225.0	2035.9	4458	225.0	2035.9	9285
225.0	2260.9	7000	225.0	2260.9	4458	225.0	2260.9	9285
225.0	2485.9	7000	225.0	2485.9	4458	225.0	2485.9	9285
225.0	2710.9	7000	225.0	2710.9	4458	225.0	2710.9	9285
225.0	2935.8	7000	225.0	2935.8	4458	225.0	2935.8	9285
225.0	3160.8	7000	225.0	3160.8	4458	225.0	3160.8	9285
225.0	3385.8	7000	225.0	3385.8	4458	225.0	3385.8	9285
553.1	3938.9	7000	553.1	3938.9	4458	553.1	3938.9	9285
553.1	4492.0	7000	553.1	4492.0	4458	553.1	4492.0	9285
553.1	5045.0	7000	553.1	5045.0	4458	553.1	5045.0	9285
553.1	5598.1	7000	553.1	5598.1	4458	553.1	5598.1	9285
587.5	6185.6	7000	587.5	6185.6	4458	587.5	6185.6	9285
3280.8	9466.5	9285	3280.8	9466.5	9285	3280.8	9466.5	9285

2.3.2.1 Shear Modulus and Damping Curves

No site-specific nonlinear dynamic material properties were determined in the initial siting of Sequoyah Nuclear Plant for sedimentary rocks. The rock material over the upper 500 ft (150 m) was assumed to have behavior that could be modeled as either linear or non-linear. To represent this potential for either case in the upper 500 ft of sedimentary rock at Sequoyah Nuclear Plant, two sets of shear modulus reduction and hysteretic damping curves were used. Consistent with the SPID (EPRI, 2013a), the EPRI rock curves (model M1) were considered to be appropriate to represent the upper range nonlinearity likely in the materials at this site and linear analyses (model M2) were assumed to represent an equally plausible alternative rock response across loading level. For the linear analyses, the low strain damping from the EPRI rock curves were used as the constant damping values in the upper 500 ft (150 m). (EPRI, 2014)

2.3.2.2 Kappa

For Sequoyah Nuclear Plant, kappa estimates were determined using Section B-5.1.3.1 of the SPID (EPRI, 2013a) for a firm CEUS rock site. Kappa for a firm rock site with at least 3,000 ft (1 km) of sedimentary rock may be estimated from the average S-wave velocity over the upper 100 ft (V_{s100}) of the subsurface profile while for a site with less than 3,000 ft (1 km) of firm rock, kappa may be estimated with a Q_s of 40 below 500 ft combined with the low strain damping from the EPRI rock curves and an additional kappa of 0.006 s for the underlying hard rock. For Sequoyah Nuclear Plant, with 6,186 ft (1,885 m) of firm sedimentary rock below the SSE, kappa estimates were based on the average shear-wave velocity over the top 100 ft (30 m) of the three base-case profiles P1, P2, and P3. For the three profiles the corresponding average shear-wave velocities were: 6,000 ft/s (1,829 m/s), 3,821 ft/s (1,165 m/s), and 9,285 ft/s (2,830 m/s) with corresponding kappa estimates of 0.012 s, 0.020 s, and 0.006 s. The range in kappa about the best estimate base-case value of 0.012 s (profile P1) is roughly 1.6 and was considered to adequately reflect epistemic uncertainty in low strain damping (kappa) for the profile. Values for kappa as well as the weights used for the site response analyses are presented below in Table 2.3.2-2. (EPRI, 2014)

Table 2.3.2-2. Kappa Values and Weights Used for Site Response Analyses. (EPRI, 2014)

Velocity Profile	Kappa(s)
P1	0.012
P2	0.020
P3	0.006
	Weights
P1	0.4
P2	0.3
P3	0.3
G/G_{max} and Hysteretic Damping Curves	
M1	0.5
M2	0.5

2.3.3 Randomization of Base Case Profiles

To account for the aleatory variability in dynamic material properties that is expected to occur across a site at the scale of a typical nuclear facility, variability in the assumed shear-wave velocity profiles has been incorporated in the site response calculations. For Sequoyah Nuclear Plant, random shear wave velocity profiles were developed from the base case profiles shown in Figure 2.3.2-1. Consistent with the discussion in Appendix B of the SPID (EPRI, 2013a), the velocity randomization procedure made use of random field models which describe the statistical correlation between layering and shear wave velocity. The default randomization parameters developed in Toro (1997) for United States Geological Survey (USGS) “A” site conditions were used for this site. Thirty random velocity profiles were generated for each base case profile. These random velocity profiles were generated using a natural log standard deviation of 0.25 over the upper 50 ft and 0.15 below that depth. As specified in the SPID (EPRI, 2013a), correlation of shear wave velocity between layers was modeled using the footprint correlation model. In the correlation model, a limit of ±2 standard deviations about the median value in each layer was assumed for the limits on random velocity fluctuations. (EPRI, 2014)

2.3.4 Input Spectra

Consistent with the guidance in Appendix B of the SPID (EPRI, 2013a), input Fourier amplitude spectra were defined for a single representative earthquake magnitude (M 6.5) using two different assumptions regarding the shape of the seismic source spectrum (single-corner and double-corner). A range of 11 different input amplitudes (median PGAs) ranging from 0.01 to 1.5g were used in the site response analyses. The characteristics of the seismic source and upper crustal attenuation properties assumed for the analysis of Sequoyah Nuclear Plant were the same as those identified in Tables B-4, B-5, B-6 and B-7 of the SPID (EPRI, 2013a) as appropriate for typical CEUS sites. (EPRI, 2014)

2.3.5 Methodology

To perform the site response analyses for Sequoyah Nuclear Plant, a random vibration theory approach was employed. This process utilizes a simple, efficient approach for computing site-specific amplification functions and is consistent with existing NRC guidance and the SPID (EPRI, 2013a). The guidance contained in Appendix B of the SPID (EPRI, 2013a) on incorporating epistemic uncertainty in shear-wave velocities, kappa, non-linear dynamic properties and source spectra for plants with limited at-site information was followed for Sequoyah Nuclear Plant. (EPRI, 2014)

2.3.6 Amplification Functions

The results of the site response analysis consist of amplification factors (5%-damped pseudo-absolute response spectra) which describe the amplification (or de-amplification) of hard reference rock motion as a function of frequency and input reference rock amplitude. The amplification factors are represented in terms of a median amplification value and an associated standard deviation (sigma) for each oscillator frequency and input rock amplitude. Consistent with the SPID (EPRI, 2013a) a minimum median amplification value of 0.5 was employed in the present analysis. Figure 2.3.6-1 illustrates the median and ±1 standard deviation in the predicted amplification factors developed for the eleven loading levels parameterized by the median reference (hard rock) peak acceleration (0.01g to 1.50g) for profile P1 and EPRI rock G/G_{max} and hysteretic damping curves. The variability in the amplification factors results from variability in shear-wave velocity, depth to hard rock, and modulus reduction and hysteretic damping curves. To illustrate the effects of nonlinearity at Sequoyah Nuclear Plant, Figure 2.3.6-2 shows the corresponding amplification factors developed with linear analyses (model M2). Little difference is seen over all loading levels for structural frequencies less than about 20 Hz. Tabular data for Figure 2.3.6-1 and Figure 2.3.6-2 is provided for information only in Appendix A. (EPRI, 2014)

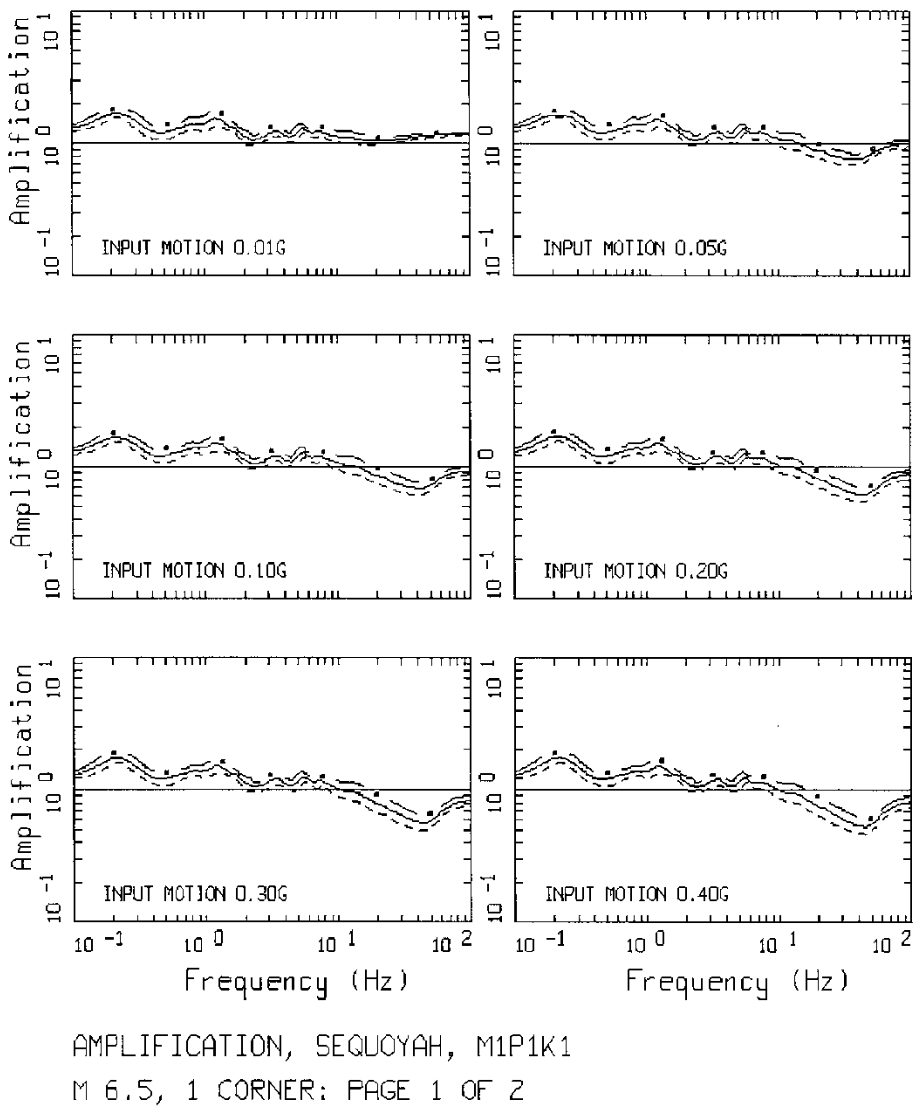


Figure 2.3.6-1. Example suite of amplification factors (5%-damping pseudo-absolute acceleration spectra) developed for the mean base-case profile (P1), EPRI rock modulus reduction and hysteretic damping curves (model M1), and base-case kappa at eleven loading levels of hard rock median peak acceleration values from 0.01g to 1.50g. M 6.5 and single-corner source model (EPRI, 2013a). (EPRI, 2014)

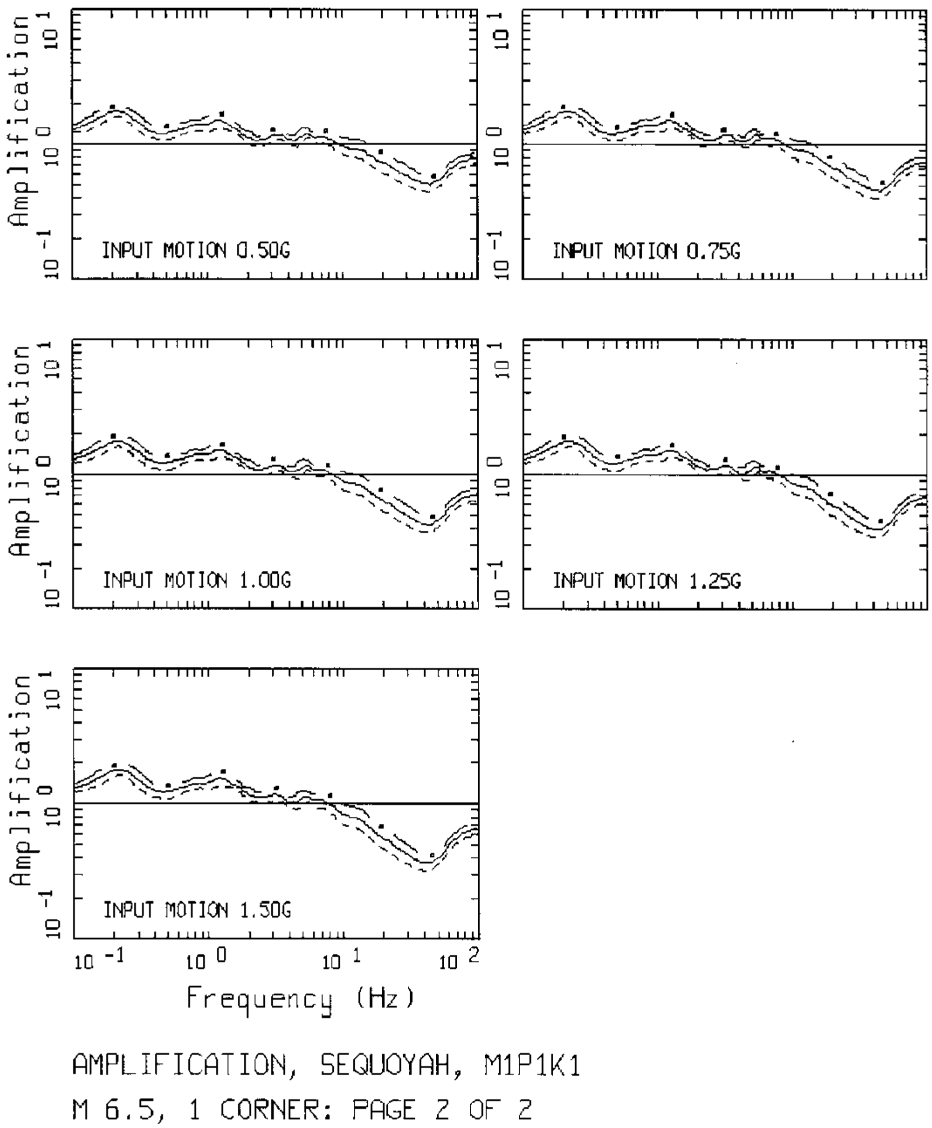


Figure 2.3.6-1.(cont.)

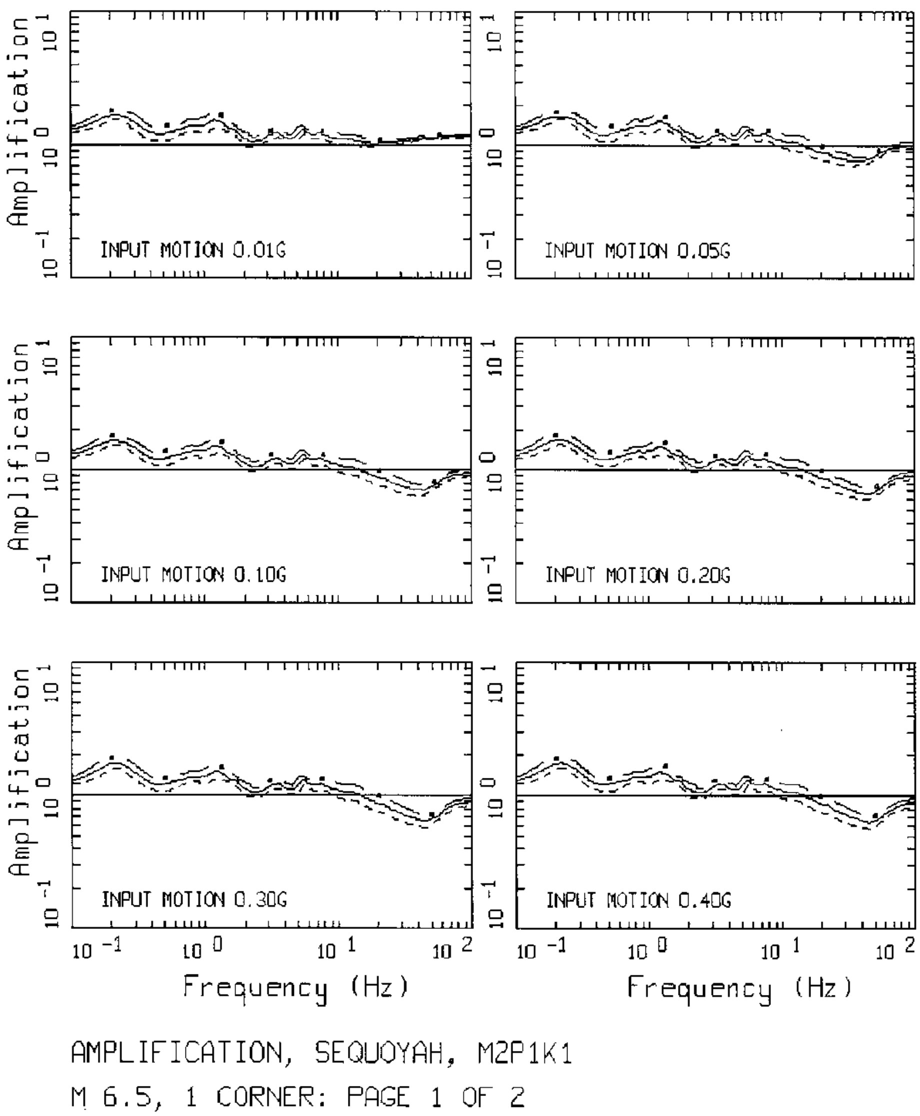
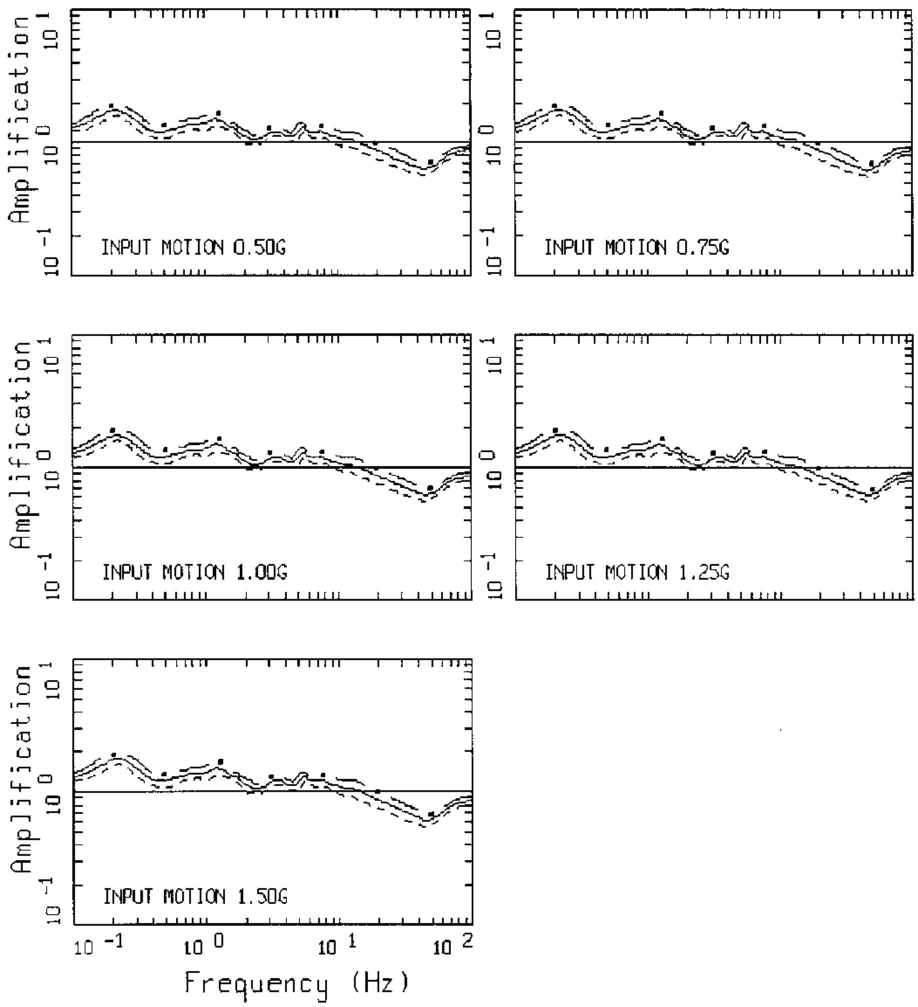


Figure 2.3.6-2. Example suite of amplification factors (5%-damping pseudo-absolute acceleration spectra) developed for the mean base-case profile (P1), linear analyses (model M2), and base-case kappa at eleven loading levels of hard rock median peak acceleration values from 0.01g to 1.50g. M 6.5 and single-corner source model (EPRI, 2013a). (EPRI, 2014)



AMPLIFICATION, SEQUOYAH, M2P1K1

M 6.5, 1 CORNER: PAGE 2 OF 2

Figure 2.3.6-2.(cont.)

2.3.7 Control Point Seismic Hazard Curves

The procedure to develop probabilistic site-specific control point hazard curves used in the present analysis follows the methodology described in Section B-6.0 of the SPID (EPRI, 2013a). This procedure (referred to as Method 3) computes a site-specific control point hazard curve for a broad range of spectral accelerations given the site-specific bedrock hazard curve and site-specific estimates of soil or soft-rock response and associated uncertainties. This process is repeated for each of the seven spectral frequencies for which ground motion equations are available. The dynamic response of the materials below the control point was represented by the frequency- and amplitude-dependent amplification functions (median values and standard deviations) developed and described in the previous section. The resulting control point mean hazard curves for Sequoyah Nuclear Plant are shown in Figure 2.3.7-1 for the seven spectral frequencies for which ground motion equations are defined. Tabulated values of mean and fractile seismic hazard curves and site response amplification functions are provided in Appendix A. (EPRI, 2014)

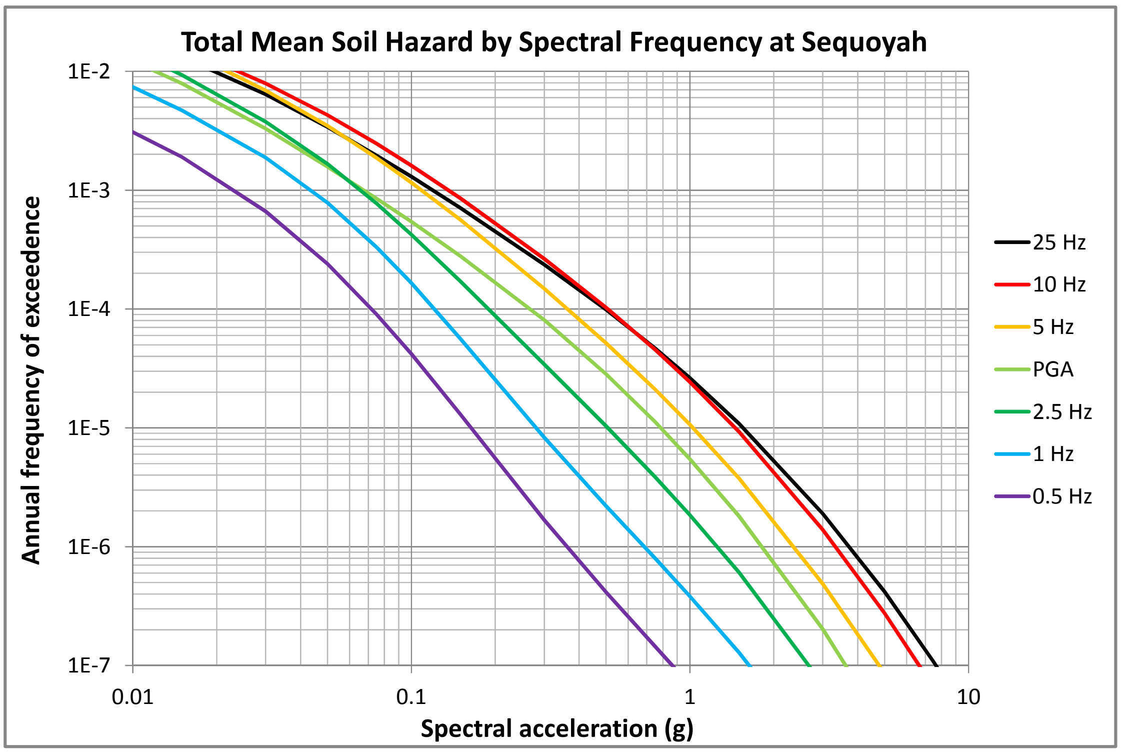


Figure 2.3.7-1. Control point mean hazard curves for spectral frequencies of 0.5, 1.0, 2.5, 5.0, 10, 25 and PGA (100 Hz) at Sequoyah Nuclear Plant. (EPRI, 2014)

2.4 Control Point Response Spectrum

The control point hazard curves described above have been used to develop Uniform Hazard Response Spectra (UHRS) and the GMRS. The UHRS were obtained through linear interpolation in log-log space to estimate the spectral acceleration at each spectral frequency for the 10^{-4} and 10^{-5} per year hazard levels.

The 10^{-4} and 10^{-5} UHRS, along with a design factor are used to compute the GMRS at the control point using the criteria in Regulatory Guide 1.208 (U.S. NRC, 2007). Table 2.4-1 shows the UHRS and GMRS spectral accelerations. (EPRI, 2014)

Table 2.4-1. UHRS and GMRS for Sequoyah Nuclear Plant. (EPRI, 2014)

Freq. (Hz)	10^{-4} UHRS (g)	10^{-5} UHRS (g)	GMRS (g)
100	2.66E-01	7.85E-01	3.79E-01
90	2.68E-01	7.94E-01	3.83E-01
80	2.70E-01	8.06E-01	3.89E-01
70	2.75E-01	8.27E-01	3.98E-01
60	2.87E-01	8.70E-01	4.18E-01
50	3.18E-01	9.70E-01	4.65E-01
40	3.76E-01	1.15E+00	5.54E-01
35	4.16E-01	1.28E+00	6.14E-01
30	4.53E-01	1.41E+00	6.72E-01
25	4.97E-01	1.55E+00	7.41E-01
20	5.19E-01	1.58E+00	7.59E-01
15	5.29E-01	1.57E+00	7.57E-01
12.5	5.29E-01	1.55E+00	7.49E-01
10	5.07E-01	1.45E+00	7.06E-01
9	4.90E-01	1.40E+00	6.82E-01
8	4.70E-01	1.34E+00	6.53E-01
7	4.41E-01	1.26E+00	6.11E-01
6	4.06E-01	1.14E+00	5.58E-01
5	3.64E-01	1.03E+00	5.00E-01
4	2.99E-01	8.29E-01	4.05E-01
3.5	2.82E-01	7.71E-01	3.78E-01
3	2.33E-01	6.38E-01	3.13E-01
2.5	1.89E-01	5.08E-01	2.50E-01
2	1.78E-01	4.63E-01	2.30E-01
1.5	1.53E-01	3.84E-01	1.92E-01
1.25	1.38E-01	3.33E-01	1.68E-01
1	1.21E-01	2.81E-01	1.42E-01
0.9	1.16E-01	2.68E-01	1.36E-01
0.8	1.07E-01	2.47E-01	1.25E-01
0.7	9.73E-02	2.24E-01	1.14E-01
0.6	8.57E-02	1.96E-01	9.98E-02
0.5	7.19E-02	1.64E-01	8.34E-02
0.4	5.75E-02	1.31E-01	6.67E-02
0.35	5.03E-02	1.15E-01	5.84E-02
0.3	4.31E-02	9.83E-02	5.00E-02
0.25	3.60E-02	8.20E-02	4.17E-02
0.2	2.88E-02	6.56E-02	3.34E-02
0.15	2.16E-02	4.92E-02	2.50E-02
0.125	1.80E-02	4.10E-02	2.08E-02
0.1	1.44E-02	3.28E-02	1.67E-02

Figure 2.4-1 shows the control point UHRS and GMRS.

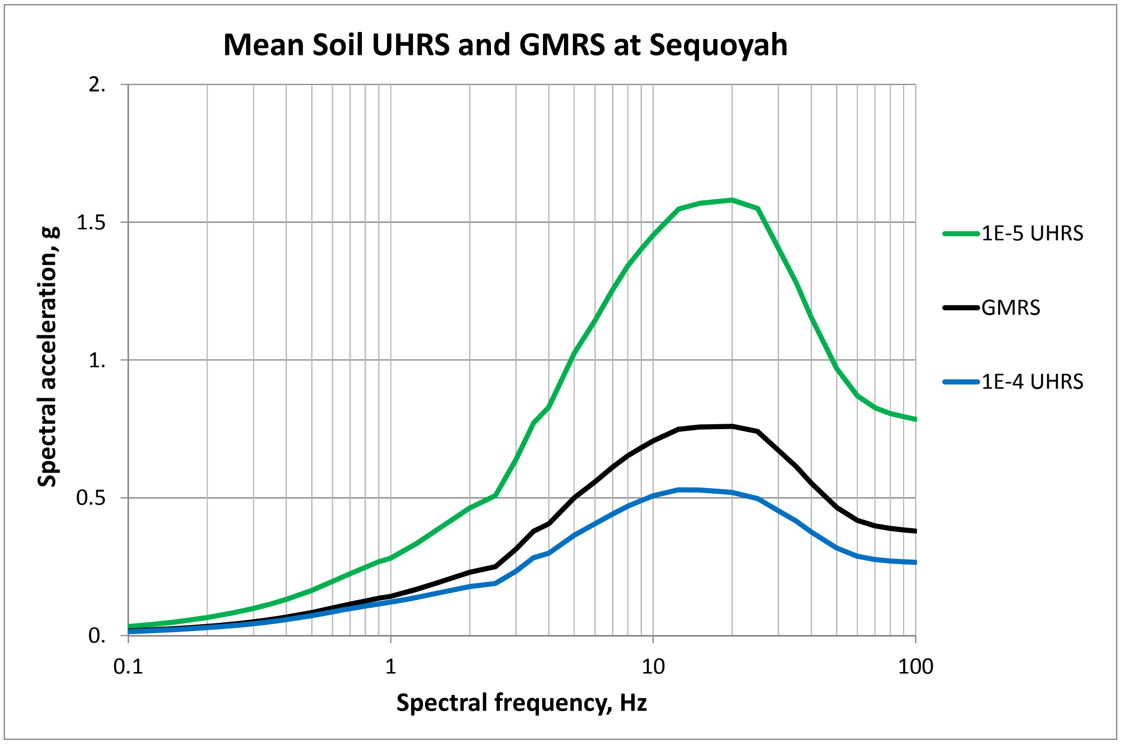


Figure 2.4-1. UHRS for 10^{-4} and 10^{-5} and GMRS at the control point for Sequoyah Nuclear Plant (5%-damped response spectra). (EPRI, 2014)

3.0 Plant Design Basis Ground Motion

The design basis for Sequoyah Nuclear Plant is identified in the Updated Final Safety Analysis Report (TVA, Amendment 24).

3.1 Safe Shutdown Earthquake Description of Spectral Shape

The SSE was developed in accordance with 10 CFR Part 100, Appendix A through an evaluation of the maximum earthquake potential for the region surrounding the site. Considering the historic seismicity of the site region, the maximum potential earthquake was determined to be an intensity VIII on the Modified Mercalli Intensity Scale of 1931 (though there is reason to believe it should have been rated as intensity VII on the Modified Mercalli Intensity Scale of 1931).

The SSE is defined in terms of a PGA and a design response spectrum. Considering a site intensity of VIII, a PGA of 0.18g was estimated. Table 3.1-1 shows the Spectral Acceleration (SA) values as a function of frequency for the 5%-damped horizontal SSE. (EPRI, 2014)

Table 3.1-1. SSE “Actual Design Spectra” for Sequoyah Nuclear Plant (Table 2; AMEC, 2013). (EPRI, 2014)

Freq. (Hz)	100	25	10	5.0	2.5	1.0	0.5
SA (g)	0.18	0.18	0.28	0.38	0.35	0.17	0.10

3.2 Control Point Elevation

The SSE control point elevation is defined at the base of the Containment Structures, which corresponds to a depth of 64 ft (Elevation 641 ft Mean Sea Level), and is the “Deepest Structure Foundation Elevation Control Point” (AMEC, 2013). (EPRI, 2014)

4.0 Screening Evaluation

In accordance with SPID (EPRI, 2013a) Section 3, a screening evaluation was performed as described below.

4.1 Risk Evaluation Screening (1 to 10 Hz)

In the 1 to 10 Hz part of the response spectrum, the GMRS exceeds the SSE. Therefore, Sequoyah Nuclear Plant screens-in for a risk evaluation.

4.2 High Frequency Screening (> 10 Hz)

For the range above 10 Hz, the GMRS exceeds the SSE. The high frequency exceedances can be addressed in the risk evaluation discussed in 4.1 above.

4.3 Spent Fuel Pool Evaluation Screening (1 to 10 Hz)

In the 1 to 10 Hz part of the response spectrum, the GMRS exceeds the SSE. Therefore, Sequoyah Nuclear Plant screens-in for a spent fuel pool evaluation.

5.0 Interim Actions

Based on the screening evaluation, the expedited seismic evaluation described in EPRI 3002000704 (EPRI, 2013c) will be performed as proposed in a letter to NRC (ML13101A379) dated April 9, 2013 (NEI, 2013) and agreed to by NRC (ML13106A331) in a letter dated May 7, 2013 (U.S. NRC, 2013).

Consistent with NRC letter (ML14030A046) dated February 20, 2014, (U.S. NRC, 2014a) the seismic hazard reevaluations presented herein are distinct from the current design and licensing bases of Sequoyah Nuclear Plant. Therefore, the results do not call into question the operability or functionality of SSCs and are not reportable pursuant to 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors," and 10 CFR 50.73, "Licensee event report system.”

The NRC letter also requests that licensees provide an interim evaluation or actions to demonstrate that the plant can cope with the reevaluated hazard while the expedited approach and risk evaluations are conducted. In response to that request, Nuclear Energy Institute (NEI) letter dated March 12, 2014 (NEI, 2014), provides seismic core damage risk estimates using the updated seismic hazards for the operating nuclear plants in the Central and Eastern United States. These risk estimates continue to support the following conclusions of the NRC GI-199 Safety/Risk Assessment (U.S. NRC, 2010):

> Overall seismic core damage risk estimates are consistent with the Commission's Safety Goal Policy Statement because they are within the subsidiary objective of 10^{-4}/year for core damage frequency. The GI-199 Safety/Risk Assessment, based in part on information from the U.S. NRC's Individual Plant Examination of External Events (IPEEE) program, indicates that no concern exists regarding adequate protection and that the current seismic design of operating reactors provides a safety margin to withstand potential earthquakes exceeding the original design basis.

Sequoyah Nuclear Plant is included in the March 12, 2014 risk estimates. Using the methodology described in the NEI letter (NEI, 2014), all plants were shown to be below 10^{-4}/year; thus, the above conclusions apply.

A full-scope Seismic Margins Assessment (SMA) was performed to support the IPEEE for Sequoyah Nuclear Plant Units 1 and 2. The results of the IPEEE for Sequoyah Nuclear Plant Units 1 and 2 were submitted to the NRC (TVA, 1995) (TVA, 2000). Results of the NRC review are documented in the referenced SER (U.S. NRC, 2001). As described in the referenced SER (U.S. NRC, 2001) TVA provided an estimated High Confidence of Low Probability of Failure (HCLPF) plant capacity of 0.23g.

Sequoyah Nuclear Plant Units 1 and 2 Seismic IPEEE was performed using the Seismic Margins Assessment option per the methodology of EPRI NP-6041-SLR1 (EPRI, 1991). With this method, a Seismic Margins Earthquake (SME) was postulated and the items needed for safe shutdown were then evaluated for the SME demand in two success paths (EPRI, 1991). Components and structures that were determined to have sufficient capacity to survive the SME without loss of function were screened out. Items that did not screen were subject to a more detailed evaluation, including calculation of a HCLPF capacity PGA for that item.

A re-assessment of the Sequoyah Nuclear Plant HCLPF capacity was performed in conjunction with development of TVA's response to NRC 10 CFR part 50.54(f) Request for information Regarding Recommendation 2.1 of the Near-Term Task Force review of insights from the Fukushima Dia-ichi accident (U.S. NRC, 2012). At the completion of the re-assessment, it was determined that a HCLPF capacity of 0.35g, defined at rock outcrop, can be achieved at Sequoyah Nuclear Plant. This improved HCLPF capacity was achieved as a result of modifications identified by the original IPEEE program and more refined analyses and seismic upgrade modifications performed subsequent to the original IPEEE reviews as shown in Table

5.0-1. In addition, an IPEEE adequacy review consistent with the requirements of Section 3.3 of the SPID (EPRI, 2013a) was performed and found that the IPEEE results at 0.35g are adequate for screening and that the risk insights gained from the IPEEE remain valid under the current plant configuration. TVA intends to submit this additional information in a subsequent letter to support the NRC staff's review of Sequoyah Nuclear Plant's risk evaluation approach and priority for completion.

In accordance with the Near-Term Task Force Recommendation 2.3 Seismic (U.S. NRC, 2012) Sequoyah Nuclear Plant Units 1 and 2 performed seismic walkdowns using the guidance in EPRI Report 1025286 (EPRI, 2012). The seismic walkdowns were completed and captured in the seismic walkdown reports (TVA, 2012) (TVA, 2014). At Sequoyah Nuclear Plant Unit 1 a total of 120 equipment items and at Sequoyah Nuclear Power Plant Unit 2 a total of 119 equipment items, were selected from the IPEEE Safe Shutdown Equipment List (SSEL) to fulfill the requirements of the seismic walkdown guidance (TVA, 2012). The selected items were located in various environments and included many different types of equipment from multiple safety systems. The walkdowns also verified that any vulnerabilities identified in Section 7.0 of the IPEEE reports (TVA, 1995) (TVA, 2000) were adequately addressed.

Twelve potentially adverse seismic conditions were identified for Unit 1 and ten potentially adverse seismic conditions were identified for Unit 2 and entered into the TVA Corrective Action Program (CAP) (TVA, 2014) (TVA, 2012). The identified potentially adverse conditions were evaluated and were found to have no operability or reportability impact on the plant. All potentially adverse seismic conditions identified for Sequoyah Nuclear Plant Units 1 and 2 have been resolved (TVA, 2013). Based on the NRC Staff's review of the seismic walkdown reports, the NRC Staff concluded that Sequoyah Nuclear Plant's implementation of the seismic walkdown methodology meets the intent of the walkdown guidance and that no immediate safety concerns were identified (U.S. NRC, 2014b, U.S. NRC 2014c).

The seismic walkdowns (TVA, 2012) (TVA, 2014) also verified in Section 7.0 that any vulnerabilities identified were adequately addressed. The seismic walkdown reports state that all of the outliers or vulnerabilities identified during the IPEEE program have been resolved either through physical modification or by refined calculations and have minimum HCLPF Capacities above 0.3g. (TVA, 2012).

Based on the NRC Staff's review of Section 7.0 of the seismic walkdown reports, the NRC Staff concluded that Sequoyah Nuclear Plant's identification of plant-specific vulnerabilities (including anomalies, outliers, and other findings) identified by the IPEEE program, as well as actions taken to eliminate or reduce them, met the intent of IPEEE vulnerabilities resolution (U.S. NRC, 2014b) (U.S. NRC, 2014c).

Table 5.0-1. IPEEE Issues and Resolutions for Sequoyah Nuclear Plants.

Equipment Name	Upgrade Applied
Residual Heat Removal Heat Exchangers	Support frame tabs to anchor plates and calculation updated to achieve 0.35g HCLPF
Main Control Room Air Handling Units	Calculation updated to achieve 0.52g HCLPF
Ice Condenser	Calculation updated to achieve 0.36g HCLPF
125 V Vital Battery Chargers	Calculation updated to achieve 0.41g HCLPF
480 V Shutdown Transformers	Anchorage replaced and calculation updated to achieve 0.47g HCLPF
480 V Shutdown Boards	Calculation updated to achieve 0.53g HCLPF
6.9 kV Shutdown Boards	Calculation updated to achieve 0.37g HCLPF
Regenerative Heat Exchangers	Calculation updated to achieve 0.48g HCLPF
480 V Diesel Auxiliary Boards	Anchorage and calculation updated to achieve 0.51g HCLPF
480 V Reactor Motor Operated Valve Boards	Calculation updated to achieve 0.51g HCLPF
480 V Control & Auxiliary Building Vent Boards	Calculation updated to achieve 0.51g HCLPF
480 V Reactor Vent Boards	Calculation updated to achieve 0.51g HCLPF
Residual Heat Removal Pumps	Calculation updated to achieve 0.42g HCLPF
120 VAC Spare Vital Inverters	Modified equipment and calculation updated to achieve 0.37g HCLPF
120 VAC U1/U2 Vital Inverters	Replaced equipment and calculation updated to achieve 0.37g HCLPF
Pipe Chase Coolers	Repaired equipment and calculation updated to achieve 0.42g HCLPF
480 V Essential Raw Cooling Water Motor Control Center	Anchorage and calculation updated to achieve 0.35g HCLPF
480 V Electric Board Room Air Handling Unit	Anchorage and calculation updated to achieve 0.57g HCLPF
Containment Spray Pump Room Cooler	Anchorage and calculation updated to achieve 0.40g HCLPF
Component Cooling Water Pumps	Calculation updated to achieve 0.57g HCLPF
6.9 kV Logic Relay Panels	Calculation updated to achieve 0.57g HCLPF
Containment Spray Pumps	Calculation updated to achieve 0.57g HCLPF
Essential Raw Cooling Water Screen Wash Pumps	Calculation updated to achieve 0.57g HCLPF
Essential Raw Cooling Water Transformers	Calculation updated to achieve 0.57g HCLPF
Residual Heat Removal Pump Room Coolers	Anchorage and calculation updated to achieve 0.40g HCLPF
Safety Injection System Pump Room Coolers	Calculation updated to achieve 0.40g HCLPF
Reciprocal Charging Pump Room Cooler	Calculation updated to achieve 0.40g HCLPF
Centrifugal Charging Pump Room Coolers	Calculation updated to achieve 0.40g HCLPF
Penetration Room Coolers	Calculation updated to achieve 0.40g HCLPF

Table 5.0-1. IPEEE Issues and Resolutions for Sequoyah Nuclear Plants. Continued.

Equipment Name	Upgrade Applied
Refueling Water Storage Tank	Calculation updated to achieve 0.40g HCLPF
Soil Failures	Calculation updated to achieve 0.35g HCLPF
120V AC Vital Instrument Power Board	Calculation updated to achieve 0.57g HCLPF
125V DC Vital Battery Inverter	Calculation updated to achieve 0.57g HCLPF
480V Board Room Supply Air Handling Unit	Calculation updated to achieve 0.41g HCLPF
480V Electrical Board Room Air Cooled Condenser	Calculation updated to achieve 0.41g HCLPF
Boric Acid Transfer Pump and Auxiliary Feedwater Pump Space Cooler	Calculation updated to achieve 0.57g HCLPF
Component Cooling Water Pump and Auxiliary Feedwater Pump Cooler	Calculation updated to achieve 0.57g HCLPF
Component Cooling Heat Exchanger	Calculation updated to achieve 0.44g HCLPF
Shutdown Board Room Supply Air Handling Unit	Calculation updated to achieve 0.45g HCLPF
Shutdown Board Room Chilled Water System Circulating Pump	Calculation updated to achieve 0.57g HCLPF
Shutdown Board Room Chiller	Calculation updated to achieve 0.57g HCLPF
Spent Fuel Pit Pump and Thermal Barrier Booster Pump Room Cooler	Calculation updated to achieve 0.68g HCLPF

6.0 Conclusions

In accordance with the 50.54(f) request for information (U.S. NRC, 2012), a seismic hazard and screening evaluation was performed for Sequoyah Nuclear Plant. A GMRS was developed solely for purpose of screening for additional evaluations in accordance with the SPID (EPRI, 2013a). Based on the results of the screening evaluation, Sequoyah Nuclear Plant screens-in for a risk evaluation, a Spent Fuel Pool evaluation, and a High Frequency Confirmation.

7.0 References

10 CFR Part 50. Title 10, Code of Federal Regulations, Part 50, “Domestic Licensing of Production and Utilization Facilities,” U.S. Nuclear Regulatory Commission, Washington DC.

10 CFR Part 100. Title 10, Code of Federal Regulations, Part 100, “Reactor Site Criteria,” U.S. Nuclear Regulatory Commission, Washington DC.

10 CFR Part 50.72. Title 10, Code of Federal Regulations, Part 50.72, “Immediate Notification Requirements for Operating Nuclear Power Reactors,” U.S. Nuclear Regulatory Commission, Washington DC.

10 CFR Part 50.73. Title 10, Code of Federal Regulations, Part 50.73, “Licensee Event Report System,” U.S. Nuclear Regulatory Commission, Washington DC.

AMEC (2013). “Seismic Data Retrieval Information for EPRI Near-Term Task Force Recommendation 2.1 TVA Sequoyah Nuclear Plant Soddy Daisy, Tennessee, Letter report,” AMEC Project 3043132002, Letter from K. Campbell to J. Best dated June 26, 2013.

CEUS-SSC (2012). “Central and Eastern United States Seismic Source Characterization for Nuclear Facilities,” U.S. Nuclear Regulatory Commission Report, NUREG-2115; EPRI Report 1021097, 6 Volumes; DOE Report# DOE/NE-0140.

Drahovzal and Neathery (1969). “Stratigraphic Succession Along the Appalachian Structural Front in Alabama.”

EPRI (1991). “A Methodology for Assessment of Nuclear Power Plant Seismic Margin,” Revision 1, NP-6041-SLR1, August 1991.

EPRI (2012). “Seismic Walkdown Guidance: For Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic,” Electric Power Research Institute, Report 1025286, June 4, 2012.

EPRI (2013a). “Seismic Evaluation Guidance Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” Electric Power Research Institute Report 1025287, February 2013.

EPRI (2013b). “EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project,” Electric Power Research Institute, Palo Alto, CA, Report 3002000717, 2 volumes, June 2013.

EPRI (2013c). EPRI 3002000704, “Seismic Evaluation Guidance, Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” May 2013.

EPRI (2014). “Sequoyah Seismic Hazard and Screening Report,” Electric Power Research Institute, Palo Alto, CA, dated February 28, 2014.

Lemiszki, Kohl, and Sutton (2008). “Geologic Map of the Decatur Quadrangle.”

NEI (2013). NEI Letter to NRC from A. Pietrangelo to D. Skeen, “Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations,” April 9, 2013.

NEI (2014). NEI Letter to NRC from A. Pietrangelo to E. Leeds, “Seismic Risk Evaluations for Plants in the Central and Eastern United States,” March 12, 2014.

Toro (1997). Appendix of: Silva, W.J., Abrahamson, N., Toro, G., and Costantino, C. (1997). "Description and Validation of the Stochastic Ground Motion Model," Report Submitted to Brookhaven National Laboratory, Associated Universities, Inc., Upton, New York 11973, Contract No. 770573.

TVA (1995). Letter from R.H. Shell to NRC "Sequoyah Nuclear Plant (SQN) - Generic Letter (GL) 88-20, Supplement No. 4, Individual Plant Evaluations of External Events I(PEEE) for Severe Accident Vulnerabilities," June 1995.

TVA (2000). Letter from Pedro Salas to NRC "Sequoyah Nuclear plant (SQN) - Units 1 and 2 - Response to Request for Additional information on the Individual Plant Examination of External Events (IPEEE) (TAC Nos. M83674 and M83675)," December 2000.

TVA (2012). Letter from J. Shea to NRC, "Tennessee Valley Authority (TVA) – Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding the Sequoyah Nuclear Plant Seismic Walkdown Results of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," November 27, 2012.

TVA (2013). Letter from J. Shea to NRC, "Revised Seismic Walkdown Reports Regarding the Seismic Aspects of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," October 4, 2013..

TVA (2014). Letter from J. Shea to NRC, "Highlights of Improvements to the Sequoyah Nuclear Plant IPEEE Seismic Analysis Results and Supplemental Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding the Sequoyah Nuclear Plant, Unit 1 Seismic Walkdown Results of Recommendation 2.3 of the Near-Term Task Force Review of Insights from Fukushima Dai-ichi Accident," January 31, 2014.

TVA (Amendment 24). Tennessee Valley Authority, "Sequoyah Nuclear Plant Updated Final Safety Analysis Report," Amendment 24.

U.S. NRC (1978). "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," NUREG/CR-0098, May 1978.

U.S. NRC (2001). Letter from NRC to J. A. Scalice, "Sequoyah Nuclear Plant, Units 1 and 2 - Review of Sequoyah Individual Pant Examination of External Events Submittal (TAC Nos. M83674 and M83675)," February 21, 2001.

U.S. NRC (2007). "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," U.S. Nuclear Regulatory Commission Reg. Guide 1.208, March 2007.

U.S. NRC (2010). "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," GI-199, September 2, 2010.

U.S. NRC (2012). NRC (E Leeds and M Johnson) Letter to All Power Reactor Licensees et al., "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," March 12, 2012.

U.S. NRC (2013). NRC Letter, Eric J. Leeds to Joseph E. Pollock, NEI "Electric Power Research Institute Final Draft Report XXXXXX, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, As an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluation," dated May 7, 2013.

U.S. NRC (2014a). NRC Letter, Eric J. Leeds to All Power Reactor Licensees, “Supplemental Information Related to Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident,” dated February 20, 2014

U.S. NRC (2014b). “Sequoyah Nuclear Plant, Unit 1- Staff Assessment of the Seismic Walkdown Report Supporting Implementation of Near-Tern Task Force Recommendation 2.3 Related to the Fukushima Dai-Ichi Nuclear Power Plant Accident (TAC NO. MF0176),” ML14022A067, January 28, 2014.

U.S. NRC (2014c). “Sequoyah Nuclear Plant, Unit 2- Staff Assessment of the Seismic Walkdown Report Supporting Implementation of Near-Tern Task Force Recommendation 2.3 Related to the Fukushima Dai-Ichi Nuclear Power Plant Accident (TAC NO. MF0177),” ML14016A039, February 6, 2014.

Appendix A

Tabulated Data

Table A-1a. Mean and Fractile Seismic Hazard Curves for 0.5 Hz at Sequoyah Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	4.15E-02	2.13E-02	2.92E-02	4.07E-02	5.35E-02	6.36E-02
0.001	2.47E-02	1.18E-02	1.64E-02	2.35E-02	3.28E-02	4.13E-02
0.005	6.17E-03	1.90E-03	3.14E-03	5.66E-03	9.24E-03	1.23E-02
0.01	3.08E-03	5.42E-04	1.10E-03	2.64E-03	5.12E-03	7.13E-03
0.015	1.90E-03	2.19E-04	4.90E-04	1.46E-03	3.37E-03	5.05E-03
0.03	6.64E-04	3.47E-05	8.98E-05	3.68E-04	1.23E-03	2.32E-03
0.05	2.39E-04	7.55E-06	2.04E-05	9.51E-05	4.13E-04	9.65E-04
0.075	9.04E-05	2.07E-06	5.66E-06	2.80E-05	1.42E-04	3.84E-04
0.1	4.18E-05	7.77E-07	2.22E-06	1.13E-05	6.26E-05	1.82E-04
0.15	1.30E-05	1.84E-07	5.75E-07	3.01E-06	1.77E-05	5.75E-05
0.3	1.68E-06	1.15E-08	4.56E-08	3.01E-07	2.16E-06	7.77E-06
0.5	4.15E-07	1.23E-09	5.75E-09	5.05E-08	4.63E-07	2.01E-06
0.75	1.45E-07	2.60E-10	1.02E-09	1.11E-08	1.31E-07	7.03E-07
1.	6.86E-08	1.51E-10	3.42E-10	3.57E-09	5.05E-08	3.19E-07
1.5	2.29E-08	1.32E-10	1.46E-10	7.03E-10	1.23E-08	9.51E-08
3.	2.87E-09	9.11E-11	1.05E-10	1.42E-10	8.47E-10	8.98E-09
5.	5.09E-10	9.11E-11	1.01E-10	1.42E-10	1.77E-10	1.25E-09
7.5	1.12E-10	9.11E-11	1.01E-10	1.42E-10	1.42E-10	2.88E-10
10.	3.51E-11	9.11E-11	9.11E-11	1.42E-10	1.42E-10	1.57E-10

Table A-1b. Mean and Fractile Seismic Hazard Curves for 1.0 Hz at Sequoyah Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	8.31E-02	4.50E-02	5.83E-02	8.35E-02	9.93E-02	9.93E-02
0.001	5.69E-02	2.60E-02	3.63E-02	5.58E-02	7.66E-02	8.98E-02
0.005	1.50E-02	5.83E-03	8.60E-03	1.40E-02	2.13E-02	2.72E-02
0.01	7.38E-03	2.39E-03	3.79E-03	6.83E-03	1.08E-02	1.44E-02
0.015	4.69E-03	1.20E-03	2.10E-03	4.19E-03	7.23E-03	9.79E-03
0.03	1.88E-03	2.76E-04	5.50E-04	1.46E-03	3.23E-03	4.83E-03
0.05	7.83E-04	7.34E-05	1.60E-04	5.05E-04	1.40E-03	2.42E-03
0.075	3.31E-04	2.35E-05	5.35E-05	1.82E-04	5.83E-04	1.15E-03
0.1	1.65E-04	9.93E-06	2.32E-05	8.12E-05	2.80E-04	5.91E-04
0.15	5.62E-05	2.88E-06	6.93E-06	2.57E-05	9.11E-05	2.10E-04
0.3	8.30E-06	3.05E-07	8.12E-07	3.37E-06	1.34E-05	3.28E-05
0.5	2.21E-06	4.63E-08	1.49E-07	7.55E-07	3.52E-06	9.11E-06
0.75	7.98E-07	8.98E-09	3.42E-08	2.19E-07	1.21E-06	3.47E-06
1.	3.83E-07	2.57E-09	1.11E-08	8.60E-08	5.58E-07	1.69E-06
1.5	1.29E-07	4.63E-10	1.98E-09	1.98E-08	1.67E-07	5.91E-07
3.	1.62E-08	1.42E-10	1.74E-10	1.20E-09	1.51E-08	7.03E-08
5.	2.86E-09	1.01E-10	1.36E-10	2.01E-10	1.92E-09	1.10E-08
7.5	6.27E-10	9.11E-11	1.01E-10	1.42E-10	3.79E-10	2.07E-09
10.	1.96E-10	9.11E-11	1.01E-10	1.42E-10	1.74E-10	6.36E-10

Table A-1c. Mean and Fractile Seismic Hazard Curves for 2.5 Hz at Sequoyah Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.14E-01	8.72E-02	9.79E-02	9.93E-02	9.93E-02	9.93E-02
0.001	9.20E-02	6.00E-02	7.23E-02	9.11E-02	9.93E-02	9.93E-02
0.005	3.04E-02	1.57E-02	2.07E-02	2.92E-02	4.07E-02	4.83E-02
0.01	1.49E-02	7.23E-03	9.79E-03	1.42E-02	2.01E-02	2.46E-02
0.015	9.30E-03	4.19E-03	5.75E-03	8.85E-03	1.29E-02	1.60E-02
0.03	3.76E-03	1.23E-03	1.87E-03	3.37E-03	5.66E-03	7.55E-03
0.05	1.67E-03	4.01E-04	6.54E-04	1.34E-03	2.68E-03	4.01E-03
0.075	7.74E-04	1.51E-04	2.57E-04	5.66E-04	1.27E-03	2.13E-03
0.1	4.24E-04	7.34E-05	1.27E-04	2.92E-04	6.93E-04	1.23E-03
0.15	1.71E-04	2.68E-05	4.70E-05	1.11E-04	2.80E-04	5.05E-04
0.3	3.43E-05	4.56E-06	8.47E-06	2.19E-05	5.66E-05	1.07E-04
0.5	1.04E-05	1.07E-06	2.19E-06	6.17E-06	1.74E-05	3.42E-05
0.75	3.87E-06	2.84E-07	6.64E-07	2.13E-06	6.54E-06	1.34E-05
1.	1.85E-06	1.01E-07	2.57E-07	9.37E-07	3.14E-06	6.73E-06
1.5	6.07E-07	1.98E-08	5.91E-08	2.64E-07	1.04E-06	2.32E-06
3.	6.96E-08	8.35E-10	2.96E-09	1.98E-08	1.10E-07	2.96E-07
5.	1.09E-08	1.57E-10	3.05E-10	2.01E-09	1.49E-08	4.83E-08
7.5	2.12E-09	1.10E-10	1.42E-10	3.42E-10	2.42E-09	9.37E-09
10.	6.03E-10	9.65E-11	1.25E-10	1.60E-10	6.54E-10	2.60E-09

Table A-1d. Mean and Fractile Seismic Hazard Curves for 5.0 Hz at Sequoyah Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.21E-01	9.79E-02	9.93E-02	9.93E-02	9.93E-02	9.93E-02
0.001	1.05E-01	7.34E-02	8.72E-02	9.93E-02	9.93E-02	9.93E-02
0.005	4.35E-02	2.22E-02	3.01E-02	4.25E-02	5.75E-02	6.54E-02
0.01	2.33E-02	1.15E-02	1.55E-02	2.25E-02	3.14E-02	3.68E-02
0.015	1.54E-02	7.23E-03	9.93E-03	1.49E-02	2.07E-02	2.53E-02
0.03	6.92E-03	2.64E-03	3.95E-03	6.45E-03	9.93E-03	1.27E-02
0.05	3.48E-03	1.05E-03	1.64E-03	3.05E-03	5.35E-03	7.34E-03
0.075	1.87E-03	4.77E-04	7.55E-04	1.51E-03	2.96E-03	4.43E-03
0.1	1.16E-03	2.64E-04	4.19E-04	8.85E-04	1.87E-03	2.96E-03
0.15	5.63E-04	1.16E-04	1.84E-04	4.01E-04	9.24E-04	1.53E-03
0.3	1.49E-04	2.84E-05	4.56E-05	1.02E-04	2.53E-04	4.19E-04
0.5	5.18E-05	9.24E-06	1.55E-05	3.52E-05	8.85E-05	1.49E-04
0.75	2.11E-05	3.37E-06	6.00E-06	1.42E-05	3.57E-05	6.17E-05
1.	1.07E-05	1.51E-06	2.80E-06	6.93E-06	1.82E-05	3.23E-05
1.5	3.76E-06	4.13E-07	8.47E-07	2.32E-06	6.36E-06	1.20E-05
3.	4.89E-07	2.72E-08	6.83E-08	2.46E-07	8.35E-07	1.77E-06
5.	8.58E-08	2.39E-09	7.03E-09	3.33E-08	1.40E-07	3.37E-07
7.5	1.84E-08	3.42E-10	9.51E-10	5.35E-09	2.76E-08	7.66E-08
10.	5.66E-09	1.55E-10	2.72E-10	1.31E-09	7.89E-09	2.46E-08

Table A-1e. Mean and Fractile Seismic Hazard Curves for 10 Hz at Sequoyah Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.19E-01	9.51E-02	9.93E-02	9.93E-02	9.93E-02	9.93E-02
0.001	1.02E-01	7.03E-02	8.72E-02	9.93E-02	9.93E-02	9.93E-02
0.005	4.15E-02	2.29E-02	3.05E-02	4.07E-02	5.05E-02	6.45E-02
0.01	2.33E-02	1.21E-02	1.62E-02	2.25E-02	2.92E-02	4.01E-02
0.015	1.61E-02	7.89E-03	1.07E-02	1.55E-02	2.07E-02	2.88E-02
0.03	7.92E-03	3.14E-03	4.50E-03	7.23E-03	1.11E-02	1.53E-02
0.05	4.28E-03	1.38E-03	2.04E-03	3.73E-03	6.54E-03	9.24E-03
0.075	2.46E-03	6.64E-04	9.93E-04	2.01E-03	3.95E-03	5.91E-03
0.1	1.61E-03	3.95E-04	5.83E-04	1.25E-03	2.64E-03	4.07E-03
0.15	8.55E-04	1.87E-04	2.80E-04	6.26E-04	1.42E-03	2.25E-03
0.3	2.64E-04	5.27E-05	8.23E-05	1.82E-04	4.56E-04	7.23E-04
0.5	1.03E-04	1.98E-05	3.23E-05	7.03E-05	1.77E-04	2.84E-04
0.75	4.56E-05	8.12E-06	1.38E-05	3.09E-05	7.77E-05	1.29E-04
1.	2.43E-05	3.95E-06	6.93E-06	1.64E-05	4.13E-05	7.03E-05
1.5	9.28E-06	1.20E-06	2.32E-06	6.00E-06	1.60E-05	2.84E-05
3.	1.39E-06	9.11E-08	2.10E-07	7.55E-07	2.42E-06	4.83E-06
5.	2.76E-07	7.89E-09	2.25E-08	1.16E-07	4.77E-07	1.07E-06
7.5	6.57E-08	8.47E-10	2.92E-09	2.13E-08	1.08E-07	2.72E-07
10.	2.17E-08	2.25E-10	6.54E-10	5.66E-09	3.37E-08	9.51E-08

Table A-1f. Mean and Fractile Seismic Hazard Curves for 25 Hz at Sequoyah Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.09E-01	6.54E-02	9.65E-02	9.93E-02	9.93E-02	9.93E-02
0.001	8.70E-02	4.50E-02	7.34E-02	8.72E-02	9.93E-02	9.93E-02
0.005	3.23E-02	1.60E-02	2.32E-02	3.09E-02	3.84E-02	6.36E-02
0.01	1.88E-02	8.72E-03	1.23E-02	1.77E-02	2.32E-02	4.01E-02
0.015	1.32E-02	5.50E-03	8.00E-03	1.21E-02	1.72E-02	2.88E-02
0.03	6.40E-03	2.04E-03	3.05E-03	5.50E-03	9.37E-03	1.46E-02
0.05	3.40E-03	8.60E-04	1.31E-03	2.72E-03	5.35E-03	8.60E-03
0.075	1.96E-03	4.07E-04	6.45E-04	1.42E-03	3.19E-03	5.35E-03
0.1	1.30E-03	2.46E-04	3.90E-04	8.98E-04	2.16E-03	3.68E-03
0.15	7.09E-04	1.23E-04	2.01E-04	4.70E-04	1.18E-03	2.07E-03
0.3	2.37E-04	3.90E-05	6.54E-05	1.55E-04	4.01E-04	7.03E-04
0.5	9.90E-05	1.49E-05	2.60E-05	6.45E-05	1.72E-04	3.01E-04
0.75	4.67E-05	6.17E-06	1.11E-05	2.96E-05	8.12E-05	1.46E-04
1.	2.63E-05	3.05E-06	5.75E-06	1.64E-05	4.50E-05	8.47E-05
1.5	1.09E-05	1.01E-06	2.04E-06	6.45E-06	1.87E-05	3.63E-05
3.	1.89E-06	9.65E-08	2.35E-07	9.11E-07	3.28E-06	6.93E-06
5.	4.17E-07	1.10E-08	3.28E-08	1.55E-07	7.03E-07	1.67E-06
7.5	1.08E-07	1.55E-09	5.35E-09	2.96E-08	1.74E-07	4.56E-07
10.	3.77E-08	4.01E-10	1.32E-09	8.23E-09	5.83E-08	1.64E-07

Table A-1g. Mean and Fractile Seismic Hazard Curves for 100 Hz (PGA) at Sequoyah Nuclear Plant. (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.00E-01	4.98E-02	8.47E-02	9.93E-02	9.93E-02	9.93E-02
0.001	7.46E-02	3.28E-02	5.91E-02	7.34E-02	9.37E-02	9.93E-02
0.005	2.30E-02	1.04E-02	1.53E-02	2.16E-02	2.84E-02	4.70E-02
0.01	1.22E-02	4.70E-03	7.03E-03	1.10E-02	1.60E-02	2.80E-02
0.015	7.92E-03	2.57E-03	3.95E-03	6.83E-03	1.10E-02	1.95E-02
0.03	3.29E-03	7.66E-04	1.16E-03	2.53E-03	5.20E-03	9.37E-03
0.05	1.57E-03	2.96E-04	4.37E-04	1.08E-03	2.57E-03	4.90E-03
0.075	8.52E-04	1.44E-04	2.19E-04	5.42E-04	1.38E-03	2.64E-03
0.1	5.42E-04	8.98E-05	1.38E-04	3.33E-04	8.85E-04	1.69E-03
0.15	2.79E-04	4.63E-05	7.13E-05	1.69E-04	4.63E-04	8.60E-04
0.3	8.06E-05	1.18E-05	1.98E-05	4.83E-05	1.32E-04	2.49E-04
0.5	2.84E-05	3.09E-06	6.09E-06	1.64E-05	4.77E-05	9.24E-05
0.75	1.12E-05	8.35E-07	1.90E-06	6.09E-06	1.87E-05	3.79E-05
1.	5.46E-06	2.72E-07	7.23E-07	2.76E-06	9.24E-06	1.92E-05
1.5	1.80E-06	4.13E-08	1.49E-07	7.66E-07	3.05E-06	6.93E-06
3.	2.03E-07	8.12E-10	5.20E-09	5.20E-08	3.14E-07	8.72E-07
5.	3.01E-08	1.42E-10	3.42E-10	4.56E-09	3.95E-08	1.38E-07
7.5	5.37E-09	1.01E-10	1.42E-10	5.66E-10	6.09E-09	2.49E-08
10.	1.40E-09	9.24E-11	1.31E-10	1.90E-10	1.42E-09	6.54E-09

Table A-2. Amplification Functions for Sequoyah Nuclear Plant. (EPRI, 2014)

PGA	Median AF	Sigma ln(AF)	25 Hz	Median AF	Sigma ln(AF)	10 Hz	Median AF	Sigma ln(AF)	5.0 Hz	Median AF	Sigma ln(AF)
1.00E-02	1.12E+00	4.00E-02	1.30E-02	1.01E+00	5.17E-02	1.90E-02	1.07E+00	8.55E-02	2.09E-02	1.19E+00	1.04E-01
4.95E-02	9.75E-01	5.44E-02	1.02E-01	8.23E-01	9.44E-02	9.99E-02	1.03E+00	1.02E-01	8.24E-02	1.17E+00	1.07E-01
9.64E-02	9.25E-01	5.93E-02	2.13E-01	7.90E-01	1.03E-01	1.85E-01	1.02E+00	1.05E-01	1.44E-01	1.17E+00	1.08E-01
1.94E-01	8.84E-01	6.30E-02	4.43E-01	7.65E-01	1.08E-01	3.56E-01	1.00E+00	1.07E-01	2.65E-01	1.16E+00	1.09E-01
2.92E-01	8.62E-01	6.50E-02	6.76E-01	7.49E-01	1.09E-01	5.23E-01	9.92E-01	1.09E-01	3.84E-01	1.15E+00	1.09E-01
3.91E-01	8.47E-01	6.62E-02	9.09E-01	7.38E-01	1.10E-01	6.90E-01	9.82E-01	1.10E-01	5.02E-01	1.14E+00	1.10E-01
4.93E-01	8.36E-01	6.70E-02	1.15E+00	7.28E-01	1.11E-01	8.61E-01	9.73E-01	1.11E-01	6.22E-01	1.14E+00	1.10E-01
7.41E-01	8.16E-01	6.77E-02	1.73E+00	7.09E-01	1.11E-01	1.27E+00	9.55E-01	1.13E-01	9.13E-01	1.12E+00	1.11E-01
1.01E+00	8.01E-01	6.77E-02	2.36E+00	6.95E-01	1.11E-01	1.72E+00	9.40E-01	1.15E-01	1.22E+00	1.11E+00	1.12E-01
1.28E+00	7.89E-01	6.67E-02	3.01E+00	6.82E-01	1.10E-01	2.17E+00	9.26E-01	1.16E-01	1.54E+00	1.10E+00	1.13E-01
1.55E+00	7.80E-01	6.64E-02	3.63E+00	6.72E-01	1.09E-01	2.61E+00	9.14E-01	1.17E-01	1.85E+00	1.09E+00	1.13E-01
2.5 Hz	**Median AF**	**Sigma ln(AF)**	**1.0 Hz**	**Median AF**	**Sigma ln(AF)**	**0.5 Hz**	**Median AF**	**Sigma ln(AF)**			
2.18E-02	1.11E+00	7.30E-02	1.27E-02	1.39E+00	1.39E-01	8.25E-03	1.26E+00	1.22E-01			
7.05E-02	1.10E+00	7.19E-02	3.43E-02	1.38E+00	1.35E-01	1.96E-02	1.26E+00	1.18E-01			
1.18E-01	1.10E+00	7.11E-02	5.51E-02	1.38E+00	1.34E-01	3.02E-02	1.26E+00	1.17E-01			
2.12E-01	1.09E+00	7.04E-02	9.63E-02	1.37E+00	1.32E-01	5.11E-02	1.26E+00	1.16E-01			
3.04E-01	1.09E+00	7.02E-02	1.36E-01	1.37E+00	1.32E-01	7.10E-02	1.26E+00	1.16E-01			
3.94E-01	1.09E+00	7.05E-02	1.75E-01	1.37E+00	1.31E-01	9.06E-02	1.26E+00	1.15E-01			
4.86E-01	1.09E+00	7.09E-02	2.14E-01	1.38E+00	1.31E-01	1.10E-01	1.26E+00	1.15E-01			
7.09E-01	1.08E+00	7.26E-02	3.10E-01	1.38E+00	1.30E-01	1.58E-01	1.26E+00	1.15E-01			
9.47E-01	1.08E+00	7.41E-02	4.12E-01	1.38E+00	1.30E-01	2.09E-01	1.26E+00	1.16E-01			
1.19E+00	1.08E+00	7.55E-02	5.18E-01	1.38E+00	1.30E-01	2.62E-01	1.26E+00	1.16E-01			
1.43E+00	1.08E+00	7.57E-02	6.19E-01	1.38E+00	1.30E-01	3.12E-01	1.27E+00	1.16E-01			

Tables A-3a and A-3b are tabular versions of the typical amplification factors provided in Figures 2.3.6-1 and 2.3.6-2. Values are provided for two input motion levels at approximately 10^{-4} and 10^{-5} mean annual frequency of exceedance. These factors are unverified and are provided for information only. The figures should be considered the governing information.

Table A-3a. Median AFs and sigmas for Model 1, Profile 1, for 2 PGA levels.(EPRI, 2014)

For Information Only

M1P1K1	Rock PGA=0.292			M1P1K1	PGA=1.01		
Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)	Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)
100.0	0.245	0.837	0.084	100.0	0.714	0.710	0.098
87.1	0.247	0.822	0.085	87.1	0.720	0.692	0.099
75.9	0.251	0.795	0.086	75.9	0.728	0.661	0.101
66.1	0.258	0.743	0.090	66.1	0.744	0.603	0.105
57.5	0.273	0.664	0.097	57.5	0.775	0.520	0.113
50.1	0.299	0.601	0.109	50.1	0.831	0.456	0.126
43.7	0.333	0.566	0.122	43.7	0.908	0.421	0.141
38.0	0.377	0.585	0.135	38.0	1.006	0.432	0.149
33.1	0.416	0.614	0.147	33.1	1.117	0.461	0.169
28.8	0.443	0.657	0.159	28.8	1.198	0.503	0.179
25.1	0.469	0.694	0.157	25.1	1.279	0.542	0.183
21.9	0.486	0.760	0.160	21.9	1.340	0.608	0.180
19.1	0.500	0.796	0.154	19.1	1.389	0.649	0.178
16.6	0.515	0.859	0.161	16.6	1.442	0.712	0.174
14.5	0.531	0.931	0.157	14.5	1.519	0.796	0.174
12.6	0.538	0.973	0.153	12.6	1.552	0.846	0.174
11.0	0.528	0.984	0.143	11.0	1.534	0.867	0.171
9.5	0.531	1.039	0.126	9.5	1.547	0.925	0.156
8.3	0.517	1.100	0.098	8.3	1.542	1.010	0.121
7.2	0.511	1.164	0.102	7.2	1.527	1.078	0.110
6.3	0.471	1.147	0.083	6.3	1.423	1.078	0.100
5.5	0.486	1.241	0.079	5.5	1.444	1.155	0.077
4.8	0.425	1.114	0.125	4.8	1.314	1.082	0.138
4.2	0.401	1.085	0.085	4.2	1.217	1.041	0.089
3.6	0.422	1.176	0.085	3.6	1.265	1.119	0.086
3.2	0.398	1.179	0.103	3.2	1.238	1.170	0.111
2.8	0.350	1.094	0.087	2.8	1.112	1.114	0.098
2.4	0.311	1.057	0.074	2.4	0.992	1.083	0.073
2.1	0.291	1.091	0.113	2.1	0.921	1.112	0.111
1.8	0.286	1.201	0.078	1.8	0.897	1.217	0.077
1.6	0.269	1.306	0.083	1.6	0.842	1.324	0.087
1.4	0.253	1.428	0.104	1.4	0.785	1.443	0.099
1.2	0.234	1.500	0.139	1.2	0.721	1.515	0.138
1.0	0.200	1.423	0.137	1.0	0.613	1.438	0.137
0.91	0.175	1.375	0.100	0.91	0.534	1.387	0.099
0.79	0.158	1.374	0.102	0.79	0.478	1.382	0.100
0.69	0.135	1.318	0.087	0.69	0.404	1.326	0.085
0.60	0.112	1.267	0.114	0.60	0.335	1.274	0.113
0.52	0.092	1.222	0.122	0.52	0.273	1.227	0.121
0.46	0.075	1.196	0.101	0.46	0.221	1.200	0.102
0.10	0.003	1.307	0.057	0.10	0.010	1.304	0.063

Table A-3b. Median AFs and sigmas for Model 2, Profile 1, for 2 PGA levels.(EPRI, 2014)

For Information Only

M2P1K1		PGA=0.292		M2P1K1		PGA=1.01	
Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)	Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)
100.0	0.262	0.897	0.067	100.0	0.871	0.867	0.071
87.1	0.265	0.883	0.067	87.1	0.883	0.850	0.071
75.9	0.270	0.856	0.068	75.9	0.904	0.820	0.072
66.1	0.280	0.807	0.068	66.1	0.944	0.765	0.072
57.5	0.300	0.730	0.071	57.5	1.025	0.687	0.076
50.1	0.336	0.674	0.079	50.1	1.165	0.639	0.085
43.7	0.380	0.645	0.097	43.7	1.334	0.620	0.104
38.0	0.434	0.673	0.114	38.0	1.530	0.657	0.122
33.1	0.476	0.702	0.121	33.1	1.672	0.691	0.126
28.8	0.503	0.745	0.135	28.8	1.752	0.736	0.140
25.1	0.529	0.783	0.135	25.1	1.828	0.775	0.139
21.9	0.543	0.850	0.144	21.9	1.861	0.843	0.147
19.1	0.554	0.882	0.138	19.1	1.878	0.877	0.141
16.6	0.565	0.942	0.146	16.6	1.901	0.938	0.148
14.5	0.575	1.008	0.142	14.5	1.919	1.005	0.143
12.6	0.581	1.051	0.134	12.6	1.924	1.049	0.136
11.0	0.567	1.056	0.126	11.0	1.863	1.053	0.127
9.5	0.565	1.107	0.120	9.5	1.846	1.104	0.121
8.3	0.542	1.153	0.092	8.3	1.757	1.151	0.092
7.2	0.533	1.214	0.098	7.2	1.718	1.212	0.098
6.3	0.490	1.191	0.082	6.3	1.570	1.190	0.083
5.5	0.504	1.289	0.068	5.5	1.609	1.287	0.068
4.8	0.434	1.136	0.112	4.8	1.377	1.134	0.112
4.2	0.411	1.112	0.087	4.2	1.299	1.111	0.087
3.6	0.433	1.205	0.081	3.6	1.361	1.204	0.081
3.2	0.401	1.189	0.087	3.2	1.257	1.188	0.087
2.8	0.350	1.094	0.078	2.8	1.092	1.094	0.078
2.4	0.311	1.055	0.074	2.4	0.966	1.055	0.074
2.1	0.291	1.089	0.114	2.1	0.902	1.089	0.113
1.8	0.286	1.201	0.082	1.8	0.884	1.200	0.081
1.6	0.269	1.304	0.080	1.6	0.827	1.301	0.080
1.4	0.253	1.426	0.106	1.4	0.774	1.423	0.105
1.2	0.233	1.498	0.138	1.2	0.711	1.494	0.137
1.0	0.199	1.420	0.137	1.0	0.604	1.417	0.135
0.91	0.175	1.373	0.100	0.91	0.528	1.370	0.099
0.79	0.158	1.372	0.102	0.79	0.473	1.370	0.101
0.69	0.134	1.317	0.087	0.69	0.401	1.316	0.086
0.60	0.112	1.267	0.114	0.60	0.333	1.266	0.113
0.52	0.092	1.221	0.121	0.52	0.271	1.221	0.120
0.46	0.075	1.196	0.101	0.46	0.220	1.195	0.100
0.10	0.003	1.307	0.057	0.10	0.010	1.302	0.063

ENCLOSURE 4

SEISMIC HAZARD AND SCREENING REPORT FOR TENNESSEE VALLEY AUTHORITY'S WATTS BAR NUCLEAR PLANT

Seismic Hazard and Screening Report for Watts Bar Nuclear Plant

Table of Contents

1.0 Introduction

Following the accident at the Fukushima Daiichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the NRC Commission established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter that requests information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements. Depending on the comparison between the reevaluated seismic hazard and the current design basis, the result is either no further risk evaluation or the performance of a seismic risk assessment. Risk assessment approaches acceptable to the staff include a seismic probabilistic risk assessment (SPRA), or a seismic margin assessment (SMA). Based upon the risk assessment results, the NRC staff will determine whether additional regulatory actions are necessary.

This report provides the information requested in items (1) through (7) of the "Requested Information" section and Attachment 1 of the 50.54(f) letter pertaining to NTTF Recommendation 2.1 for the Watts Bar Nuclear Plant, located in Rhea County, Tennessee. In providing this information, Tennessee Valley Authority has followed the guidance provided in the *Seismic Evaluation Guidance: Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic* (EPRI 1025287, 2012). The Augmented Approach, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic (EPRI 3002000704, 2013), has been developed as the process for evaluating critical plant equipment as an interim action to demonstrate additional plant safety margin, prior to performing the complete plant seismic risk evaluations.

The original geologic and seismic siting investigations for Watts Bar Nuclear Plant were performed in accordance with Appendix A to 10 CFR Part 100 and meet General Design Criterion 2 in Appendix A to 10 CFR Part 50. The Safe Shutdown Earthquake Ground Motion (SSE) was developed in accordance with Appendix A to 10 CFR Part 100 and used for the design of seismic Category I systems, structures and components.

In response to the 50.54(f) letter and following the guidance provided in the SPID (EPRI 1025287, 2012), a seismic hazard reevaluation was performed. For screening purposes, a Ground Motion Response Spectrum (GMRS) was developed. Based on the results of the screening evaluation, the Watts Bar Nuclear Plant screens-in for a risk evaluation, a Spent Fuel Pool evaluation, and a High Frequency Confirmation.

2.0 Seismic Hazard Reevaluation

Watts Bar Nuclear Plant is located approximately 50 miles northeast of Chattanooga, Tennessee, on the west side of the Tennessee River at river mile 528 (TVA, Amendment 11, Section 2.1.1 and 2.1.3). Watts Bar Nuclear Plant is located in the Tennessee section of the Valley and Ridge Province of the Appalachian Highlands. This section is the southernmost of the three sections comprising the Valley and Ridge Province and extends from the Tennessee River-New River divide southwestward into central Alabama. It is bounded on the west by the Appalachian Plateaus Province and on the east by the Blue Ridge Province (TVA, Amendment 11, Section 2.5).

The evaluation of the earthquake hazard at the Watts Bar Nuclear Plant site involves the consideration of the seismic history not only of the immediate area but of the entire southeast and adjacent areas. The most seismically active areas are described in the following summary (TVA, Amendment 11, Section 2.5.2.1).

a. The Upper Mississippi Embayment, especially the New Madrid region of Arkansas, Kentucky, Missouri, and Tennessee. A few great earthquakes and thousands of light to moderately strong shocks have been centered in the Upper Mississippi Embayment area. Light to moderate shocks are still occurring at a frequency of a few per year in this zone. This region is more than 285 miles west-northwest of the Watts Bar Nuclear Plant site.
b. The Lower Wabash Valley of Illinois and Indiana. This area has been the focus of several moderately strong earthquakes. The effects at the Watts Bar Nuclear Plant site from future shocks epicentered in this area would be greatly attenuated as the mouth of the Wabash River is 235 miles to the northwest.
c. South Carolina Area. There is an apparent zone of seismic activity extending from Charleston, South Carolina, on the southeast northwestward across the Piedmont. One of the country's greatest earthquakes occurred near Charleston in 1886. Minor to moderate shocks have occurred subsequently along this alignment. Charleston is 285 miles southeast of the Watts Bar Nuclear site.
d. Southern Appalachian Tectonic Province. This zone extends from central Virginia to central Alabama from the western edge of the Piedmont across the Cumberland Plateau. The Watts Bar Nuclear Plant site lies within this province which is a region of continuing minor earthquake activity. Light to moderate shocks occur at an average frequency of one or two per year. The activity is not uniform, as periods of several shocks per year are followed by longer periods of no perceptible shocks.

In addition to these areas, shocks of light to moderate intensity from widely scattered epicenters have occurred at other localities in the southeastern United States at various distances from Watts Bar Nuclear Plant. The maximum historic earthquake reported in the Watts Bar Nuclear Plant site province was assigned an intensity of VIII on the Modified Mercalli Intensity Scale of 1931 and occurred in Giles County, Virginia in 1897. Even though this earthquake occurred 255 miles northeast of Watts Bar Nuclear Plant, this earthquake intensity is assumed to occur adjacent to the site for the purpose of defining the SSE. The SSE for the Watts Bar Nuclear Plant has been established as having a maximum horizontal peak ground acceleration (PGA) of 0.18g and a simultaneous maximum vertical PGA of 0.12g (TVA, Amendment 11, Section 2.5.2.4).

2.1 Regional and Local Geology

The Watts Bar Nuclear Plant site is located in the Tennessee section of the Valley and Ridge Province of the Appalachian Highlands. The Valley and Ridge Province is a long, narrow belt trending NE-SW that is bordered by the Appalachian Plateau on the west and by the Blue Ridge Province on the east. It extends for 1,200 miles from eastern New York to central Alabama. Its maximum width is 80 miles. The maximum width in east Tennessee is 40 miles, which is near the average for the southern half of the province. This province is made up of a series of folded and faulted mountains and valleys, which are underlain by Paleozoic sedimentary formations totaling 40,000 feet (ft.) in thickness (TVA, Amendment 11, Section 2.5.1.1.3).

Within the Valley and Ridge Province, sedimentary rocks from Pennsylvanian to Cambrian age are found with those of Cambrian and Ordovician age predominating. In Tennessee, the Rome Formation and the Conasauga, Knox, and Chickamauga Groups makeup the majority of the bedrock of the Valley and Ridge Province. They outcrop as repeated belts that trend NE-SW as the result of major Paleozoic thrust faulting from the southeast. The maximum exposed thickness of the Middle Cambrian Rome is about 1,200 to 1,500 ft. It is composed mostly of shales, siltstones, and sandstones. The Middle Cambrian Conasauga Group is mainly alternating shale and limestone along the southeastern border of the province and nearly all shale along the northwest border of the province. It is about 2,000 ft. thick and forms the bedrock for the Watts Bar Nuclear Plant. The Knox Group is 2,500 to 3,000 ft. thick and is of Late Cambrian to Early Ordovician age. It is mostly dolomite with some limestone.

The Chickamauga Group is Middle Ordovician in age and ranges in thickness from about 8,000 ft. in the southeast to 2,000 ft. in the northwest. It is mainly alternating layers of limestone, siltstone, and shale. Elsewhere in the Valley and Ridge are sandstones, shales, and limestones of Late Ordovician to Pennsylvanian age.

The geologic structure of the Valley and Ridge is characterized by numerous elongate folds and thrust faults that trend northeast-southwest. In the southern section of the province the faults, and in most places the bedding, dip southeast. These orientations are the result of folding and fracturing during a mountain building episode 230 to 260 million years ago. Approximately one mile northwest of the Watts Bar Nuclear Plant lies the Kingston fault, and about four miles to the southeast lies the Whiteoak Mountain fault. These faults are prominent members of two of the three families of faults that dominate Rodgers' “belt of dominant folding” – the Kingston, Whiteoak Mountain, and Saltville families. The Kingston fault begins in Anderson County, Tennessee and runs for about 175 miles southwest through Tennessee, across the northwest corner of Georgia and may extend into Alabama. The

Whiteoak Mountain fault begins in southwest Virginia and extends for a length of about 235 miles southwestward across Tennessee into northwest Georgia.

The highly deformed character of the Conasauga Formation at the Watts Bar Nuclear Plant site is a function of its lithology and structural history. Lithologically, the formation consists of several hundred feet of interstratified shale and limestone. In the plant site area, shale beds make up eighty-four percent of the formation and limestone beds the remaining sixteen percent. The shale strata are much less competent than the interstratified limestone strata. The general strike of the strata is North 30 degrees East, and the overall dip is to the southeast, but the many small, tightly folded, steeply pitching anticlines and synclines result in many local variations to the normal trend (TVA, Amendment 11, Section 2.5.1.2.3).

Stratigraphically, the Conasauga Formation is overlain by 2,500 to 3,000 feet of massive dolomite and limestone of the Knox Group and is underlain by 800 to 1,200 feet of sandstone and shale of the Rome Formation. Sometime in the course of the Appalachian orogeny, these formations were thrust northwestward on the Kingston thrust sheet, which overrode the underlying rocks for an undetermined distance. Before the thrusting ceased, the belt of Conasauga on which the plant site is located was compressed between the two massive blocks of the much more competent underlying Rome Formation and the overlying Knox Group. As a result of the very marked difference in competency between the limestone and shale in the Conasauga, and the much greater disparity between the competency of the Rome and Knox and that of the Conasauga, the latter was folded, contorted, crumpled, sheared, and broken by small faults (TVA, Amendment 11, Section 2.5.1.2.3).

2.2 Probabilistic Seismic Hazard Analysis

2.2.1 Probabilistic Seismic Hazard Analysis Results

In accordance with the 50.54(f) letter and following the guidance in the SPID (EPRI, 2013a), a probabilistic seismic hazard analysis (PSHA) was completed using the recently developed Central and Eastern United States Seismic Source Characterization (CEUS-SSC) for Nuclear Facilities (CEUS-SSC, 2012) together with the updated EPRI Ground-Motion Model (GMM) for the CEUS (EPRI, 2013b). For the PSHA, a lower-bound moment magnitude of 5.0 was used, as specified in the 50.54(f) letter.

For the PSHA, the CEUS-SSC background seismic sources out to a distance of 400 miles (640 km) around Watts Bar Nuclear Plant were included. This distance exceeds the 200 mile (320 km) recommendation (U. S. NRC, 2007) and was chosen for completeness. Background sources included in this site analysis were the following: (EPRI, 2014)

1. Extended Continental Crust—Atlantic Margin (ECC_AM)
2. Extended Continental Crust—Gulf Coast (ECC_GC)
3. Illinois Basin Extended Basement (IBEB)
4. Mesozoic and younger extended prior – narrow (MESE-N)
5. Mesozoic and younger extended prior – wide (MESE-W)

6. Midcontinent-Craton alternative A (MIDC_A)
7. Midcontinent-Craton alternative B (MIDC_B)
8. Midcontinent-Craton alternative C (MIDC_C)
9. Midcontinent-Craton alternative D (MIDC_D)
10. Non-Mesozoic and younger extended prior – narrow (NMESE-N)
11. Non-Mesozoic and younger extended prior – wide (NMESE-W)
12. Paleozoic Extended Crust narrow (PEZ_N)
13. Paleozoic Extended Crust wide (PEZ_W)
14. Reelfoot Rift (RR)
15. Reelfoot Rift including the Rough Creek Graben (RR-RCG)
16. Study region (STUDY_R)

For sources of large magnitude earthquakes, designated Repeated Large Magnitude Earthquake (RLME) sources in NUREG-2115 (CEUS-SSC, 2012) modeled for the CEUS-SSC, the following sources lie within 1,000 km of the site and were included in the analysis (EPRI, 2014):

1. Charleston
2. Commerce
3. Eastern Rift Margin Fault northern segment (ERM-N)
4. Eastern Rift Margin Fault southern segment (ERM-S)
5. Marianna
6. New Madrid Fault System (NMFS)
7. Wabash Valley

For each of the above background and RLME sources, the mid-continent version of the updated CEUS EPRI GMM was used.

2.2.2 Base Rock Seismic Hazard Curves

Consistent with the SPID (EPRI, 2013a), base rock seismic hazard curves are not provided as the site amplification approach referred to as Method 3 has been used. Seismic hazard curves are shown below in Figure 2.3.7-1at the SSE control point elevation (EPRI, 2014).

2.3 Site Response Evaluation

Following the guidance contained in Seismic Enclosure 1 of the 50.54(f) Request for Information (U.S. NRC, 2012) and in the SPID (EPRI, 2013a) for nuclear power plant sites that are not founded on hard rock (defined as 2.83 km/sec), a site response analysis was performed for Watts Bar Nuclear Plant (EPRI, 2014).

2.3.1 Description of Subsurface Material

Watts Bar Nuclear Plant is located in the Tennessee section of the Appalachian Valley and Ridge Physiographic Province. The province is made up of a series of folded and faulted mountains and valleys that are underlain by Paleozoic sedimentary rock totaling about 40,000 ft (12,200m) in thickness. The site is located on the northern end of the Chickamauga Reservoir in eastern Tennessee near Spring City (EPRI, 2014).

The information used to create the site geologic profile at the Watts Bar Nuclear Plant is shown in Tables 2.3.1-1 and 2.3.1-2. This profile was developed using information documented in AMEC (2013). As indicated in Table 2.3.1-1, the SSE Control Point is at a depth of 64 ft (19.5 m). The SSE control point lies on interbedded shales and limestones (Table 2.3.1-1). Tables 2.3.1-1 and 2.3.1-2 show the stratigraphic column, description, depth, best estimate shear-wave velocity and velocity range. Depth to basement below the Rome Formation is at a depth of about 10,950 ft (3,340m) (Table 2.3.1-2) (EPRI, 2014).

Table 2.3.1-1 Summary of Geotechnical Profile Data for Watts Bar Nuclear Plant (AMEC, 2013).
(EPRI, 2014)

Depth (feet)	Soil/Rock Description	Density (pcf)	Measured Vs* (fps)	Vs for Analyses (fps)	Gmax (psf)	G/Gmax vs. Shear Strain	Damping Ratio vs. Shear Strain
0	Ground Surface Elev. 728	-	-	-	-	-	-
0 - 32	In-situ Clays, Silts, Sand and Gravel**	120	700 - 1,830	1,200	6,500,000	FSAR Figure 2.5-233E	FSAR Figure 2.5-233F
32 - 64	Interbedded Shales and Limestones	165	4,160 - 8,341	5,000 - 7,000	200,000,000	1	No Change
64	**Deepest Structure Foundation Control Point – SSE GMRS**	-	-	-	-	-	-
64 - 180	Interbedded Shales and Limestones	165	4,160 - 8,341	5,000 - 7,000	200,000,000	1	No Change

Note –* The range of shear wave velocities measured in various geophysical tests performed at the site.
** Replaced with engineered backfill for safety related structures.

Table 2.3.1-2 Summary of Geotechnical Profile Data for Watts Bar Nuclear Plant Extended to Basement (AMEC, 2013). (EPRI, 2014)

Depth (feet)	Soil/Rock Description*	Rock Formation	Best Estimate V_s (fps)**	Lower Range V_s (fps)***	Upper Range V_s (fps)***
0	Shale, predominantly gray to greenish- gray, thin-bedded; siltstone, gray, thin-bedded, glauconitic and calcareous; limestone, thin-bedded, discontinuous beds. Commonly weathers to rust colors. Overlying soil is yellowish brown and commonly contains fragments of shale and siltstone.	Є cm - Conasauga Middle(Weathered Overburden)	1460	1168	1825
1.8	Shale, predominantly gray to greenish- gray, thin-bedded; siltstone, gray, thin-bedded, glauconitic and calcareous; limestone, thin-bedded, discontinuous beds. Commonly weathers to rust colors. Overlying soil is yellowish brown and commonly contains fragments of shale and siltstone.	Є cm - Conasauga Middle(Weathered Overburden)	1700	1360	2125
2.5	Shale, predominantly gray to greenish- gray, thin-bedded; siltstone, gray, thin- bedded, glauconitic and calcareous; limestone, thin-bedded, discontinuous beds. Commonly weathers to rust colors. Overlying soil is yellowish brown and commonly contains fragments of shale and siltstone.	Є cm - Conasauga Middle(Weathered Overburden)	600	480	750
3.8	Shale, predominantly gray to greenish-gray, thin-bedded; siltstone, gray, thin- bedded, glauconitic and calcareous; limestone, thin-bedded, discontinuous beds. Commonly weathers to rust colors. Overlying soil is yellowish brown and commonly contains fragments of shale and siltstone.	Є cm - Conasauga Middle(Weathered Overburden)	450	360	565
10.8	Shale, predominantly gray to greenish- gray, thin-bedded; siltstone, gray, thin- bedded, glauconitic and calcareous; limestone, thin-bedded, discontinuous beds. Commonly weathers to rust colors. Overlying soil is yellowish brown and commonly contains fragments of shale and siltstone.	Є cm - Conasauga Middle(Weathered Overburden)	1000	800	1250

Table 2.3.1-2 Summary of Geotechnical Profile Data for Watts Bar Nuclear Plant Extended to Basement(AMEC, 2013),Continued (EPRI, 2014)

33	Shale, predominantly gray to greenish- gray, thin-bedded; siltstone, gray, thin-bedded, glauconitic and calcareous; limestone, thin-bedded, discontinuous beds. Commonly weathers to rust colors. Overlying soil is yellowish brown and commonly contains fragments of shale and siltstone.	Є cm - Conasauga Middle(Weathered Overburden)	1700	1360	2125
76	Shale, predominantly gray to greenish-gray, thin-bedded; siltstone, gray, thin- bedded, glauconitic and calcareous; limestone, thin-bedded, discontinuous beds. Commonly weathers to rust colors. Overlying soil is yellowish brown and commonly contains fragments of shale and siltstone.	Є cm - Conasauga Middle	2400	1920	3000
136	Shale, predominantly gray to greenish-gray, thin-bedded; siltstone, gray, thin- bedded, glauconitic and calcareous; limestone, thin-bedded, discontinuous beds. Commonly weathers to rust colors. Overlying soil is yellowish brown and commonly contains fragments of shale and siltstone.	Є cm - Conasauga Middle	6000	4800	7500
136-656	Shale, predominantly gray to greenish-gray, thin-bedded; siltstone, gray, thin- bedded, glauconitic and calcareous; limestone, thin-bedded, discontinuous beds. Commonly weathers to rust colors. Overlying soil is yellowish brown and commonly contains fragments of shale and siltstone. Siltstone, greenish-gray, glauconitic, micaceous, very bioturbated, interbedded with fine-grained sandstone and shale.	Є cm - Conasauga Middle and Є pv – Pumpkin Valley Shale	6000	4800	7500
656-1000	Sandstone, reddish-brown, greenish- gray, light-brown, olive, fine-to medium-grained, thin-to thick-bedded, glauconitic, micaceous; interbedded with shale and siltstone, reddish- brown, olive greenish-gray, light-brown, thin-bedded, micaceous, bioturbated; dolomite and dolomoitic limestone may also be present; thrust fault at base, estimated exposed thickness shown.	Єr – Rome Formation	7750	6200	9285

Table 2.3.1-2 Summary of Geotechnical Profile Data for Watts Bar Nuclear Plant Extended to Basement(AMEC, 2013),Continued (EPRI, 2014)

1000-1400	Sandstone, reddish-brown, greenish-gray, light-brown, olive, fine-to medium-grained, thin-to thick-bedded, glauconitic, micaceous; interbedded with shale and siltstone, reddish- brown, olive greenish-gray, light- brown, thin-bedded, micaceous, bioturbated; dolomite and dolomoitic limestone may also be present; thrust fault at base, estimated exposed thickness shown.	Єr – Rome Formation	10,000	8000	9285
1400-2350	Sandstone, reddish-brown, greenish-gray, light-brown, olive, fine-to medium-grained, thin-to thick-bedded, glauconitic, micaceous; interbedded with shale and siltstone, reddish- brown, olive greenish-gray, light- brown, thin-bedded, micaceous, bioturbated; dolomite and dolomoitic limestone may also be present; thrust fault at base, estimated exposed thickness shown.	Єr – Rome Formation	10,000	6370	9285
2350		Kingston Fault			
2350-2700	Dolomite, light-gray with pinkish streaks and hues, fine-grained, thick-to massive-bedded, laminations; scattered quartz sand grains; limestone, light-gray, fine-grained; medium- to massive-bedded, thrombolitic, silicified gastropods; chert pods, light-gray, red, some oolitic; chert bedded, white and gastropods, and stromatolite. Base defined by chert matrix sandstone float.	Oma – Mascot Dolomite	7000	4460	9285
2700-2900	Dolomite, light-gray, fine- to coarse- grained, medium- to thick-bedded, rare oolites and scattered quartz sand grains; dolomite in the upper part is gray with pink streaks or pinkish hues; limestone, light- to medium gray, fine- grained, thick- to massive-bedded; base defined by chert, thick- to massive bedded, fine-grained, white, gastropods.	Ok – Kingsport Formation	7000	4460	9285
2900-3450	Dolomite, light-gray, tan, fine- to medium-grained, medium- to thick-bedded; chert, light-gray and white, pods, lenses, beds, oolitic, dolomoldic, fine-grained; base defined by sandstone float consisting of medium-grained quartz, and ripple laminations.	Oc – Chepultepec	7000	4460	9285

Table 2.3.1-2 Summary of Geotechnical Profile Data for Watts Bar Nuclear Plant Extended to Basement(AMEC, 2013),Continued (EPRI, 2014)

3450-4250	Dolomite, dark-gray, brownish-gray, medium- to coarse-grained (saccharoidal), medium- to massive- bedded, petroliferous odor when broken; dolomite, light-gray, fine- to coarse-grained, medium- to thick- bedded; chert, pods, lenses and beds, medium- to coarse-grained oolitic, cryptozoon, gray and white banded.	Єcr – Copper Ridge Dolomite	7000	4460	9285
4250-4450	Limestone, light- to medium-gray, medium- to massive-bedded, dolomite ribbons, fine-grained, oolitic, stylolites, thrombolitic; minor shale, green, thin- bedded.	Єmn – Maynardville Formation (Limestone)	9500	6050	9285
4450-6350	Shale, gray and greenish-gray, thin- bedded; siltstone, gray, thin-bedded, glauconitic and calcareous; limestone, thin-bedded, edgewise conglomerates consisting of dolomitic rip-up clasts throughout the middle and upper part of the formation. Lower part consists of interbedded siltstone, and shale, gray and greenish-gray, thin-bedded, glauconitic, micaceous, commonly bioturbated, a few marine shell fossils found.	Єcl – Conasauga Group Lower Undivided	7000	4460	9285
6350		Chattanooga Fault			
6350-6450	Dolomite, light-gray, tan, fine-to medium-grained, medium- to thick-bedded; chert, light-gray and white, pods, lenses, beds, oolitic, dolomoldic, fine-grained; base defined by sandstone float consisting of medium-grained quartz, and ripple laminations.	Oc – Chepultepec Dolomite	7000	4460	9285
6450-7200	Dolomite, dark-gray, brownish-gray, medium- to coarse-grained (saccharoidal), medium- to massive- bedded, petroliferous odor when broken; dolomite, light-gray, fine- to coarse-grained, medium- to thick- bedded; chert, pods, lenses and beds, medium- to coarse-grained oolitic, cryptozoon, gray and white banded.	Єcr – Copper Ridge Dolomite	7000	4460	9285
7200-7700	Limestone, light- to medium-gray, medium- to massive-bedded, dolomite ribbons, fine-grained, oolitic, stylolites, thrombolitic; minor shale, green, thin- bedded.	Emn – Maynardville Formation (limestone)	9500	6050	9285

Table 2.3.1-2 Summary of Geotechnical Profile Data for Watts Bar Nuclear Plant Extended to Basement(AMEC, 2013),Continued (EPRI, 2014)

7700-8450	Shale, predominantly greenish- and brownish-gray, thin-bedded; limestone, thin-bedded, edgewise conglomerates consisting of dolomitic rip-up clasts throughout the formation; limestone in the lower part is thick-bedded, glauconitic, oolitic. A thick- to massive bedded, light- to medium-gray, oolitic, thrombolitic, and ribboned limestone reef (Єnr) occurs near the middle of the formation, which may contain irregular infillings of dark-gray, granular limestone.	Єn – Nolichucky Shale	7000	4460	9285
8450-9050	Shale, predominantly gray to greenish-gray, thin-bedded; siltstone, gray, thin- bedded, glauconitic and calcareous; limestone, thin-bedded, discontinuous beds. Commonly weathers to rust colors. Overlying soil is yellowish brown and commonly contains fragments of shale and siltstone.	Єcm – Conasauga Group Middle	7000	4460	9285
9050-9450	Siltstone, greenish-gray, glauconitic, micaceous, very bioturbated, interbedded with fine-grained sandstone and shale.	Єpv – Pumpkin Valley Shale	7000	4460	9285
9450-10,600	Sandstone, reddish-brown, greenish- gray, light-brown, olive, fine-to medium-grained, thin-to thick-bedded, glauconitic, micaceous; interbedded with shale and siltstone, reddish- brown, olive greenish-gray, light- brown, thin-bedded, micaceous, bioturbated; dolomite and dolomoitic limestone may also be present; thrust fault at base, estimated exposed thickness shown.	Єr – Rome Formation	10,000	6370	9285
10,600		Sequatchie Valley Fault			
10,600-10,950	Sandstone, reddish-brown, greenish- gray, light-brown, olive, fine-to medium-grained, thin-to thick-bedded, glauconitic, micaceous; interbedded with shale and siltstone, reddish- brown, olive greenish-gray, light-brown, thin-bedded, micaceous, bioturbated; dolomite and dolomoitic limestone may also be present; thrust fault at base, estimated exposed thickness shown.	Єr – Rome Formation	10,000	6370	9285
>10,950		Basement	12,000	7640	9285

*Note: Rock Descriptions obtained from the Lemiski et al. (2008).
**Note: For depths of 0–1400 ft, these values were based on SASW testing by Dr. Ken Stokoe. For depths of 1400 ft to basement, these values were inferred based both on the previous SASW testing and collaboration with Ivan Wong from URS, who assisted Dr. Stokoe and AMEC in developing a lognormal average for the best estimate.
***Note: The lower and upper ranges were based on the best estimate, with the upper range constrained not to exceed 9285 fps. For depths of 0–1400 ft, these values were calculated using a certainty of 1.25. For depths of 1400 ft to basement, these values were calculated using a certainty of 1.57.
****Note: The top of the Rome Formation can vary between 656 feet and 1000 feet deep based on the dip of the strata beneath the site. Thus, the range of 656-1000 feet is shown in these layers.

The following description of the Paleozoic sequence is taken directly from (AMEC, 2013) (EPRI, 2014):

> "The Watts Bar Nuclear Plant site is underlain by the Conasauga Group of Cambrian Age which consists of mainly interbedded shale and limestone with predominant shale at the subject site. In this area, the Conasauga Shale consists of light green and dull purple shale with thin light blue lenses of limestone. It is about 2,000 feet thick (Tennessee Division of Geology, 1956).
>
> "Underlying the Conasauga Group is the Rome Formation which consists mainly of olive green silty shale. It also contains some sandstone in small lenses (Tennessee Division of Geology, 1956). The Rome Formation is about 1,200 to 1,500 feet thick (TVA Watts Bar Nuclear Plant, updated FSAR). According to the Decatur Geologic Quad, Tennessee Division of Geology (Lemiszki et al., 2008), the Kingston Fault underlies the site, and "daylights" approximately 3,000 feet to the west of the reactor buildings. The geologic quad indicates that the fault is dipping to the southeast at approximately 35 degrees. With interpolation, the fault runs approximately 4,000 feet beneath the reactor buildings and has resulted in rocks of Cambrian age unconformably overlying Ordovician strata.
>
> "According to the Tennessee Division of Geology Bulletin 58 (Tennessee Division of Geology, 1956), the Chickamauga consists of a cherty, silty limestone. According to the Geologic Map of Tennessee (Hardeman et al., 1966), this section of the Chickamauga is about 1,400 feet thick. Below the Chickamauga is the Knox Group which consists generally of siliceous dolomite. It can be anywhere from 2,500 to 3,000 feet thick. Based on the Geological Society of America Special Paper 433 (Hatcher et al., 2007), the sole fault and the basement rock beneath the fault is approximately 2.5 kilometers (8,200 feet) below ground surface."

2.3.2 Development of Base Case Profiles and Nonlinear Material Properties

Tables 2.3.1-1 and 2.3.1-2 and additional shear-wave velocity versus depth Figures 1 and 2 in AMEC (2013) shows the recommended shear-wave velocities along with depths and corresponding stratigraphy. From Table 2.3.1-1 the SSE control point is at a depth of 64 ft (19.5 m). The additional figures in AMEC (2013) provide detailed shear-wave velocity information for the top part of the site profile. These velocities are from recently performed Spectral-Analysis-of-Surface-Waves (SASW) measurements at the Watts Bar Nuclear Plant site. SASW was performed at multiple locations to capture the variability in shear-wave velocity across the site due to the underlying Kingston fault and velocities were measured at depths greater than 1,000 ft (305 m). These deep measurements were important to evaluate the thickness of the firm rock across the site (EPRI, 2014).

Shear-wave velocities listed on Table 2.3.1-1 were based on SASW and Birdwell velocity measurements (AMEC, 2013). Base-case as well as upper- and lower-range profiles were based on Figures 1 and 2 in AMEC (2013). The shear-wave velocity measurements extended to a depth below the SSE of greater than 1,000 ft (305m) into the Rome Formation. This formation has a measured shear-wave velocity of greater than 10,000 ft/s (3,050m/s). To accommodate a deterministic change in depth to hard rock conditions (at or exceeding 9,850 ft/s (2,830m/s)) across the site, two depths were specified in AMEC (2013) (Figures 1 and 2): 592 ft (180m) randomized ± 178ft (54m) (P1) and 936 ft (285m) randomized ±281 ft (85.6m) (P4). The depth randomization reflects ± 30% of the depth and was included to provide a realistic broadening of the fundamental resonance at deep sites in addition to reflect actual random variations in depth to basement shear-wave velocities across a footprint (EPRI, 2014).

Lower- and upper-range profiles, P2 and P3 respectively for shallow depths to hard rock conditions and P5 and P6 respectively for deeper depths to hard rock used a scale factor of 1.25 reflecting multiple measured shear-wave velocity estimates over the top 1,000 ft (305m). The scale factor of 1.25 reflect $\sigma_{\mu ln}$ of about 0.2 based on the SPID (EPRI, 2013a) 10th and 90th fractiles which implies a 1.28 scale factor on σ_{μ}. The six base-case profiles are shown in Figure 2.3.2-1 and listed in Table 2.3.2-1 and Table 2.3.2-2 (EPRI, 2014).

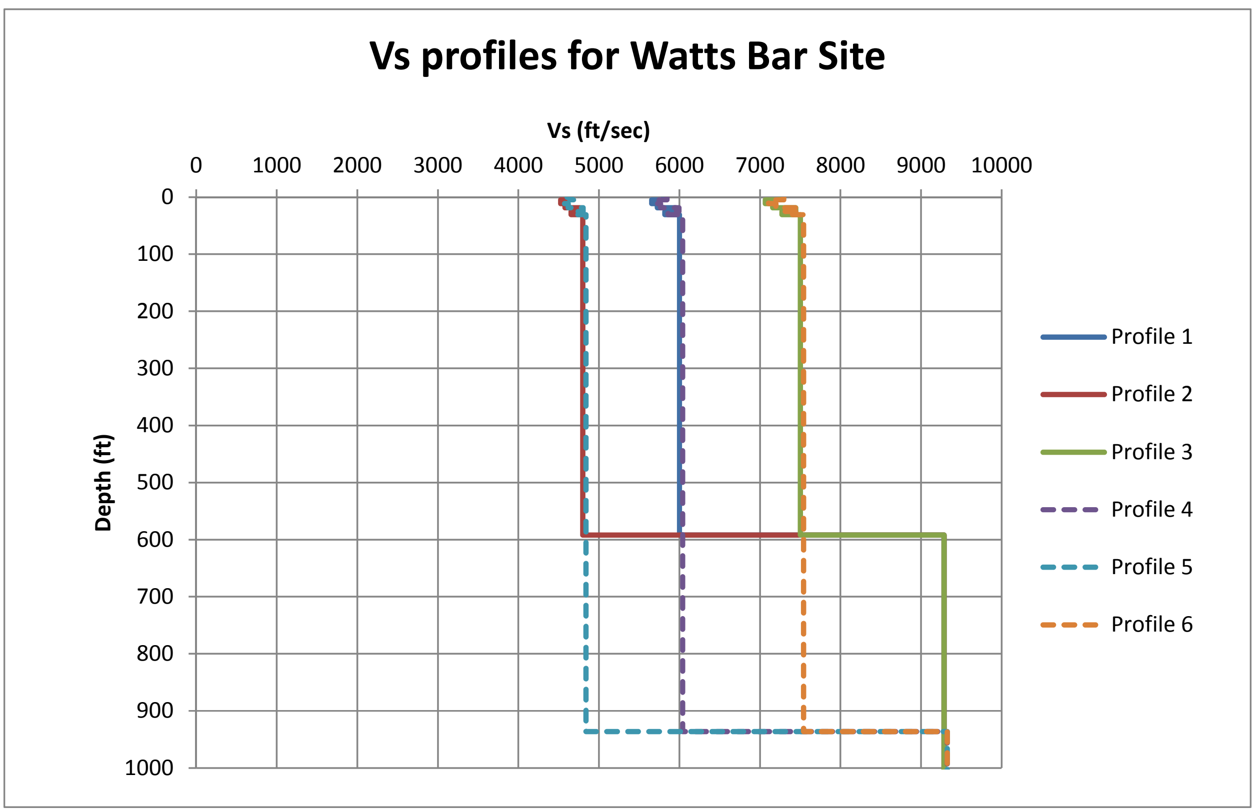


Figure 2.3.2-1. Shear-wave velocity profiles for Watts Bar Nuclear Plant . (EPRI, 2014)

Table 2.3.2-1. Layer thicknesses, depths, and shear-wave velocities (Vs) for profiles 1 to 3, Watts Bar Nuclear Plant. (EPRI, 2014)

Profile 1			Profile 2			Profile 3		
thickness(ft)	depth (ft)	Vs(ft/s)	thickness(ft)	depth (ft)	Vs(ft/s)	thickness(ft)	depth (ft)	Vs(ft/s)
	0	5805		0	4644		0	7256
5.0	5.0	5805	5.0	5.0	4644	5.0	5.0	7256
6.5	11.5	5654	6.5	11.5	4524	6.5	11.5	7068
7.0	18.5	5725	7.0	18.5	4580	7.0	18.5	7156
1.5	20.0	5954	1.5	20.0	4763	1.5	20.0	7443
5.0	25.0	5954	5.0	25.0	4763	5.0	25.0	7443
5.0	30.0	5819	5.0	30.0	4655	5.0	30.0	7274
1.0	31.0	5819	1.0	31.0	4655	1.0	31.0	7274
9.0	40.0	6000	9.0	40.0	4800	9.0	40.0	7500
10.0	50.0	6000	10.0	50.0	4800	10.0	50.0	7500
10.0	60.0	6000	10.0	60.0	4800	10.0	60.0	7500
10.0	70.0	6000	10.0	70.0	4800	10.0	70.0	7500
10.0	80.0	6000	10.0	80.0	4800	10.0	80.0	7500
10.0	90.0	6000	10.0	90.0	4800	10.0	90.0	7500
10.0	100.0	6000	10.0	100.0	4800	10.0	100.0	7500
10.0	110.0	6000	10.0	110.0	4800	10.0	110.0	7500
10.0	120.0	6000	10.0	120.0	4800	10.0	120.0	7500
10.0	130.0	6000	10.0	130.0	4800	10.0	130.0	7500
10.0	140.0	6000	10.0	140.0	4800	10.0	140.0	7500
10.0	150.0	6000	10.0	150.0	4800	10.0	150.0	7500
10.0	160.0	6000	10.0	160.0	4800	10.0	160.0	7500
10.0	170.0	6000	10.0	170.0	4800	10.0	170.0	7500
10.0	180.0	6000	10.0	180.0	4800	10.0	180.0	7500
10.0	190.0	6000	10.0	190.0	4800	10.0	190.0	7500
10.0	200.0	6000	10.0	200.0	4800	10.0	200.0	7500
10.0	210.0	6000	10.0	210.0	4800	10.0	210.0	7500
10.0	220.0	6000	10.0	220.0	4800	10.0	220.0	7500
10.0	230.0	6000	10.0	230.0	4800	10.0	230.0	7500
10.0	240.0	6000	10.0	240.0	4800	10.0	240.0	7500
10.0	250.0	6000	10.0	250.0	4800	10.0	250.0	7500
10.0	260.0	6000	10.0	260.0	4800	10.0	260.0	7500
10.0	270.0	6000	10.0	270.0	4800	10.0	270.0	7500
10.0	280.0	6000	10.0	280.0	4800	10.0	280.0	7500
10.0	290.0	6000	10.0	290.0	4800	10.0	290.0	7500
10.0	300.0	6000	10.0	300.0	4800	10.0	300.0	7500

Table 2.3.2-1. Layer thicknesses, depths, and shear-wave velocities (Vs) for profiles 1 to 3, Watts Bar Nuclear Plant , Continued. (EPRI, 2014)

10.0	310.0	6000	10.0	310.0	4800	10.0	310.0	7500
10.0	320.0	6000	10.0	320.0	4800	10.0	320.0	7500
10.0	330.0	6000	10.0	330.0	4800	10.0	330.0	7500
10.0	340.0	6000	10.0	340.0	4800	10.0	340.0	7500
10.0	350.0	6000	10.0	350.0	4800	10.0	350.0	7500
10.0	360.0	6000	10.0	360.0	4800	10.0	360.0	7500
10.0	370.0	6000	10.0	370.0	4800	10.0	370.0	7500
10.0	380.0	6000	10.0	380.0	4800	10.0	380.0	7500
10.0	390.0	6000	10.0	390.0	4800	10.0	390.0	7500
10.0	400.0	6000	10.0	400.0	4800	10.0	400.0	7500
10.0	410.0	6000	10.0	410.0	4800	10.0	410.0	7500
10.0	420.0	6000	10.0	420.0	4800	10.0	420.0	7500
10.0	430.0	6000	10.0	430.0	4800	10.0	430.0	7500
10.0	440.0	6000	10.0	440.0	4800	10.0	440.0	7500
10.0	450.0	6000	10.0	450.0	4800	10.0	450.0	7500
10.0	460.0	6000	10.0	460.0	4800	10.0	460.0	7500
10.0	470.0	6000	10.0	470.0	4800	10.0	470.0	7500
10.0	480.0	6000	10.0	480.0	4800	10.0	480.0	7500
10.0	490.0	6000	10.0	490.0	4800	10.0	490.0	7500
10.0	500.0	6000	10.0	500.0	4800	10.0	500.0	7500
18.4	518.4	6000	18.4	518.4	4800	18.4	518.4	7500
18.4	536.8	6000	18.4	536.8	4800	18.4	536.8	7500
18.4	555.2	6000	18.4	555.2	4800	18.4	555.2	7500
18.4	573.6	6000	18.4	573.6	4800	18.4	573.6	7500
18.4	592.0	6000	18.4	592.0	4800	18.4	592.0	7500
3280.8	3872.8	9285	3280.8	3872.8	9285	3280.8	3872.8	9285

Table 2.3.2-2. Layer thicknesses, depths, and shear-wave velocities (Vs) for profiles 4 to 6, Watts Bar Nuclear Plant site. (EPRI, 2014)

Profile 4			Profile 5			Profile 6		
thickness(ft)	depth (ft)	Vs(ft/s)	thickness(ft)	depth (ft)	Vs(ft/s)	thickness(ft)	depth (ft)	Vs(ft/s)
	0	5805		0	4644		0	7256
5.0	5.0	5805	5.0	5.0	4644	5.0	5.0	7256
6.5	11.5	5654	6.5	11.5	4524	6.5	11.5	7068
7.0	18.5	5725	7.0	18.5	4580	7.0	18.5	7156
1.5	20.0	5954	1.5	20.0	4763	1.5	20.0	7443
5.0	25.0	5954	5.0	25.0	4763	5.0	25.0	7443
5.0	30.0	5819	5.0	30.0	4655	5.0	30.0	7274
1.0	31.0	5819	1.0	31.0	4655	1.0	31.0	7274
9.0	40.0	6000	9.0	40.0	4800	9.0	40.0	7500
10.0	50.0	6000	10.0	50.0	4800	10.0	50.0	7500
10.0	60.0	6000	10.0	60.0	4800	10.0	60.0	7500
10.0	70.0	6000	10.0	70.0	4800	10.0	70.0	7500
10.0	80.0	6000	10.0	80.0	4800	10.0	80.0	7500
10.0	90.0	6000	10.0	90.0	4800	10.0	90.0	7500
10.0	100.0	6000	10.0	100.0	4800	10.0	100.0	7500
10.0	110.0	6000	10.0	110.0	4800	10.0	110.0	7500
10.0	120.0	6000	10.0	120.0	4800	10.0	120.0	7500
10.0	130.0	6000	10.0	130.0	4800	10.0	130.0	7500
10.0	140.0	6000	10.0	140.0	4800	10.0	140.0	7500
10.0	150.0	6000	10.0	150.0	4800	10.0	150.0	7500
10.0	160.0	6000	10.0	160.0	4800	10.0	160.0	7500
10.0	170.0	6000	10.0	170.0	4800	10.0	170.0	7500
10.0	180.0	6000	10.0	180.0	4800	10.0	180.0	7500
10.0	190.0	6000	10.0	190.0	4800	10.0	190.0	7500
10.0	200.0	6000	10.0	200.0	4800	10.0	200.0	7500
10.0	210.0	6000	10.0	210.0	4800	10.0	210.0	7500
10.0	220.0	6000	10.0	220.0	4800	10.0	220.0	7500
10.0	230.0	6000	10.0	230.0	4800	10.0	230.0	7500
10.0	240.0	6000	10.0	240.0	4800	10.0	240.0	7500
10.0	250.0	6000	10.0	250.0	4800	10.0	250.0	7500
10.0	260.0	6000	10.0	260.0	4800	10.0	260.0	7500
10.0	270.0	6000	10.0	270.0	4800	10.0	270.0	7500
10.0	280.0	6000	10.0	280.0	4800	10.0	280.0	7500
10.0	290.0	6000	10.0	290.0	4800	10.0	290.0	7500
10.0	300.0	6000	10.0	300.0	4800	10.0	300.0	7500

Table 2.3.2-2. Layer thicknesses, depths, and shear-wave velocities (Vs) for profiles 4 to 6, Watts Bar Nuclear Plant site, Continued. (EPRI, 2014)

10.0	310.0	6000	10.0	310.0	4800	10.0	310.0	7500
10.0	320.0	6000	10.0	320.0	4800	10.0	320.0	7500
10.0	330.0	6000	10.0	330.0	4800	10.0	330.0	7500
10.0	340.0	6000	10.0	340.0	4800	10.0	340.0	7500
10.0	350.0	6000	10.0	350.0	4800	10.0	350.0	7500
10.0	360.0	6000	10.0	360.0	4800	10.0	360.0	7500
10.0	370.0	6000	10.0	370.0	4800	10.0	370.0	7500
10.0	380.0	6000	10.0	380.0	4800	10.0	380.0	7500
10.0	390.0	6000	10.0	390.0	4800	10.0	390.0	7500
10.0	400.0	6000	10.0	400.0	4800	10.0	400.0	7500
10.0	410.0	6000	10.0	410.0	4800	10.0	410.0	7500
10.0	420.0	6000	10.0	420.0	4800	10.0	420.0	7500
10.0	430.0	6000	10.0	430.0	4800	10.0	430.0	7500
10.0	440.0	6000	10.0	440.0	4800	10.0	440.0	7500
10.0	450.0	6000	10.0	450.0	4800	10.0	450.0	7500
10.0	460.0	6000	10.0	460.0	4800	10.0	460.0	7500
10.0	470.0	6000	10.0	470.0	4800	10.0	470.0	7500
10.0	480.0	6000	10.0	480.0	4800	10.0	480.0	7500
10.0	490.0	6000	10.0	490.0	4800	10.0	490.0	7500
10.0	500.0	6000	10.0	500.0	4800	10.0	500.0	7500
87.2	587.2	6000	87.2	587.2	4800	87.2	587.2	7500
87.2	674.4	6000	87.2	674.4	4800	87.2	674.4	7500
87.2	761.6	6000	87.2	761.6	4800	87.2	761.6	7500
87.2	848.8	6000	87.2	848.8	4800	87.2	848.8	7500
87.2	936.0	6000	87.2	936.0	4800	87.2	936.0	7500
3280.8	4216.8	9285	3280.8	4216.8	9285	3280.8	4216.8	9285

2.3.2.1 Shear Modulus and Damping Curves

Recent site-specific nonlinear dynamic material properties were not available for sedimentary rocks at Watts Bar Nuclear Plant. The rock material over the upper 500 ft (150 m) was assumed to have behavior that could be modeled as either linear or non-linear. To represent this potential for either case in the upper 500 ft of sedimentary rock at the Watts Bar Nuclear Power Plant site, two sets of shear modulus reduction and hysteretic damping curves were used. Consistent with the SPID (EPRI, 2013a), the EPRI rock curves (model M1) were considered to be appropriate to represent the upper range nonlinearity likely in the materials at this site and linear analyses (model M2) was assumed to represent an equally plausible alternative rock response across loading level. For the linear analyses, the low strain damping from the EPRI rock curves were used as the constant damping values in the upper 500 ft (150 m) (EPRI, 2014).

2.3.2.2 Kappa

Base-case kappa estimates were determined using Section B-5.1.3.1 of the SPID (EPRI, 2013a) for a firm CEUS rock site. Kappa for a firm rock site with less than 3,000 ft (1 km) of firm rock may be estimated with a Q_s of 40 below 500 ft combined with the low strain damping from the EPRI rock curves and an additional kappa of 0.006s for the underlying hard rock. For the shallow depth to hard rock (592 ft, 180m), profiles P1, P2, and P3, the total kappa estimates (including the additional kappa of 0.006 s for the underlying hard rock) were 0.012s, 0.013s, and 0.011s respectively. For the deeper depth to hard rock (936 ft, 285m), profiles P4, P5, and P6, the total kappa estimates (including the additional kappa of 0.006 s for the underlying hard rock) were 0.013s, 0.015s, and 0.012s respectively. These values resulted in a range considered inadequate to reflect epistemic uncertainty in kappa for the site. To accommodate a larger expression of epistemic uncertainty in kappa, a scale factor of 1.68 (EPRI, 2013a) about the kappa estimate of profile P1 was used for profiles P2 and P3 resulting in estimates of 0.020s and 0.007s respectively (Table 2.3.2-3). Similarly, a scale factor of 1.68 (EPRI, 2013a) about the kappa estimate of profile P4 was used for profiles P5 and P6 resulting in estimates of 0.022s and 0.008s, respectively (Table 2.3.2-3) (EPRI, 2014).

Table 2.3.2-3
Kappa Values and Weights Used for Site Response Analyses. (EPRI, 2014)

Velocity Profile	Kappa(s)
P1	0.012
P2	0.020
P3	0.007
P4	0.013
P5	0.022
P6	0.008
	Weights
P1	0.20
P2	0.15
P3	0.15
P4	0.20
P5	0.15
P6	0.15
G/G_{max} and Hysteretic Damping Curves	Weights
M1	0.5
M2	0.5

2.3.3 Randomization of Base Case Profiles

To account for the aleatory variability in dynamic material properties that is expected to occur across a site at the scale of a typical nuclear facility, variability in the assumed shear-wave velocity profiles has been incorporated in the site response calculations. For the Watts Bar Nuclear Plant, random shear wave velocity profiles were developed from the base case profiles shown in Figure 2.3.2-1. Consistent with the discussion in Appendix B of the SPID (EPRI, 2013a), the velocity randomization procedure made use of random field models which describe the statistical correlation between layering and shear wave velocity. The default randomization parameters developed in Toro (1997) for USGS "A" site conditions were used for this site. Thirty random velocity profiles were generated for each base case profile. These random velocity profiles were generated using a natural log standard deviation of 0.25 over the upper 50 ft and 0.15 below that depth. As specified in the SPID (EPRI, 2013a), correlation of shear wave velocity between layers was modeled using the footprint correlation model. In the correlation model, a limit of +/- 2 standard deviations about the median value in each layer was assumed for the limits on random velocity fluctuations (EPRI, 2014).

2.3.4 Input Spectra

Consistent with the guidance in Appendix B of the SPID (EPRI, 2013a), input Fourier amplitude spectra were defined for a single representative earthquake magnitude (**M** 6.5) using two different assumptions regarding the shape of the seismic source spectrum (single-corner and double-corner). A range of 11 different input amplitudes (median peak ground accelerations (PGA) ranging from 0.01 to 1.5 g) were used in the site response analyses. The characteristics of the seismic source and upper crustal attenuation properties assumed for the analysis of the Watts Bar Nuclear Plant were the same as those identified in Tables B-4, B-5, B-6 and B-7 of the SPID (EPRI, 2013a) as appropriate for typical CEUS sites (EPRI, 2014).

2.3.5 Methodology

To perform the site response analyses for the Watts Bar Nuclear Plant, a random vibration theory approach was employed. This process utilizes a simple, efficient approach for computing site-specific amplification functions and is consistent with existing NRC guidance and the SPID (EPRI, 2013a). The guidance contained in Appendix B of the SPID (EPRI, 2013a) on incorporating epistemic uncertainty in shear-wave velocities, kappa, non-linear dynamic properties and source spectra for plants with limited at-site information was followed for the Watts Bar Nuclear Plant (EPRI, 2014).

2.3.6 Amplification Functions

The results of the site response analysis consist of amplification factors (5% damped pseudo absolute response spectra) which describe the amplification (or de-amplification) of hard reference rock motion as a function of frequency and input reference rock amplitude. The amplification factors are represented in terms of a median amplification value and an associated standard deviation (sigma) for each oscillator frequency and input rock amplitude. Consistent with the SPID (EPRI, 2013a) a minimum median amplification value of 0.5 was employed in the present analysis. Figure 2.3.6-1 illustrates the median and +/- 1 standard deviation in the predicted amplification factors developed for the eleven loading levels parameterized by the median reference (hard rock) peak acceleration (0.01g to 1.50g) for profile P1 and EPRI (2013a) rock G/G_{max} and hysteretic damping curves. The variability in the amplification factors results from variability in shear-wave velocity, depth to hard rock, and modulus reduction and hysteretic damping curves. To illustrate the effects of nonlinearity at the Watts Bar Nuclear Plant firm rock site, Figure 2.3.6-2 shows the corresponding amplification factors developed with linear site response analyses (model M2). Between the linear and nonlinear (equivalent-linear) analyses, Figures 2.3.6-1 and Figure 2.3.6-2 respectively show only a minor difference for frequencies below about 20 Hz and the 0.5g loading level and below. Above about the 0.5g loading level, the differences increase significantly but only above about 20 Hz. Tabular data for Figure 2.3.6-1 and Figure 2.3.6-2 is provided for information only in Appendix A (EPRI, 2014).

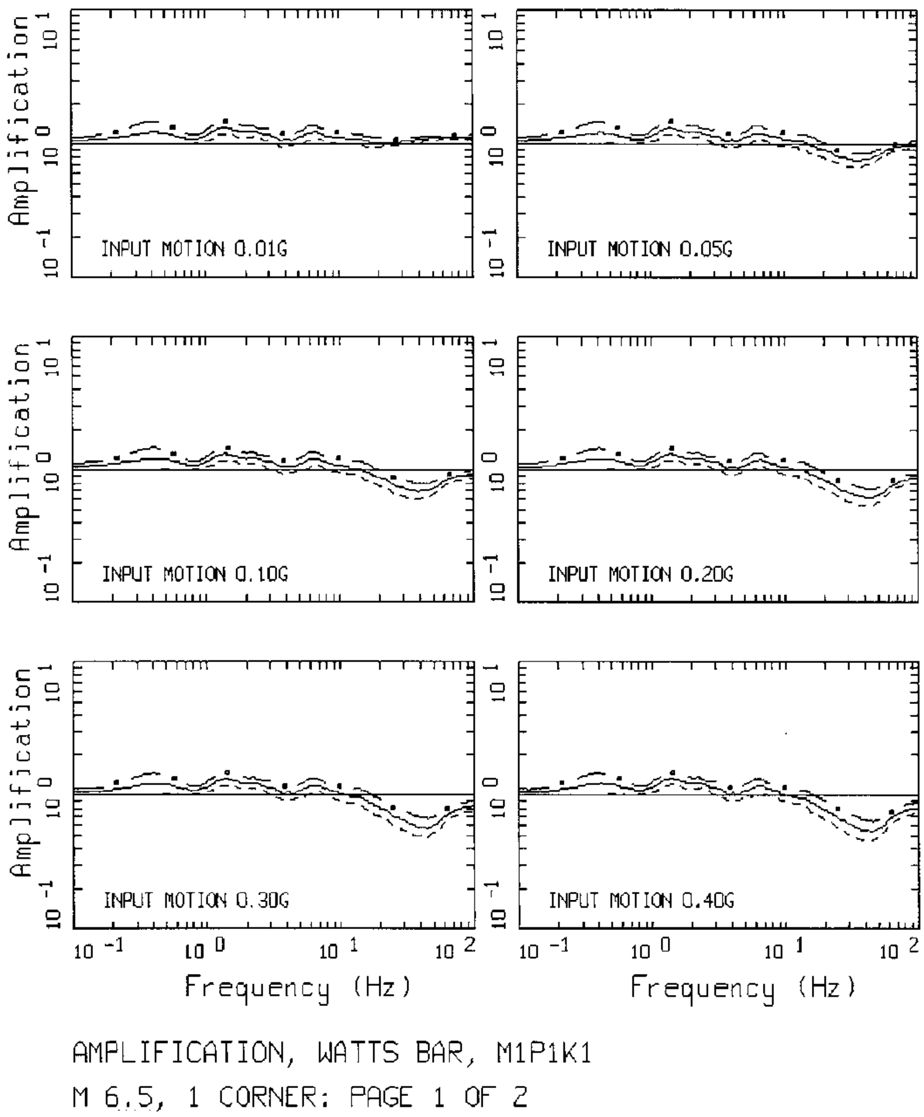


Figure 2.3.6-1. Example suite of amplification factors (5% damping pseudo absolute acceleration spectra) developed for the mean base-case profile (P1), EPRI rock modulus reduction and hysteretic damping curves (model M1), and base-case kappa (K1) at eleven loading levels of hard rock median peak acceleration values from 0.01g to 1.50g. **M** 6.5 and single-corner source model (EPRI, 2013a). (EPRI, 2014)

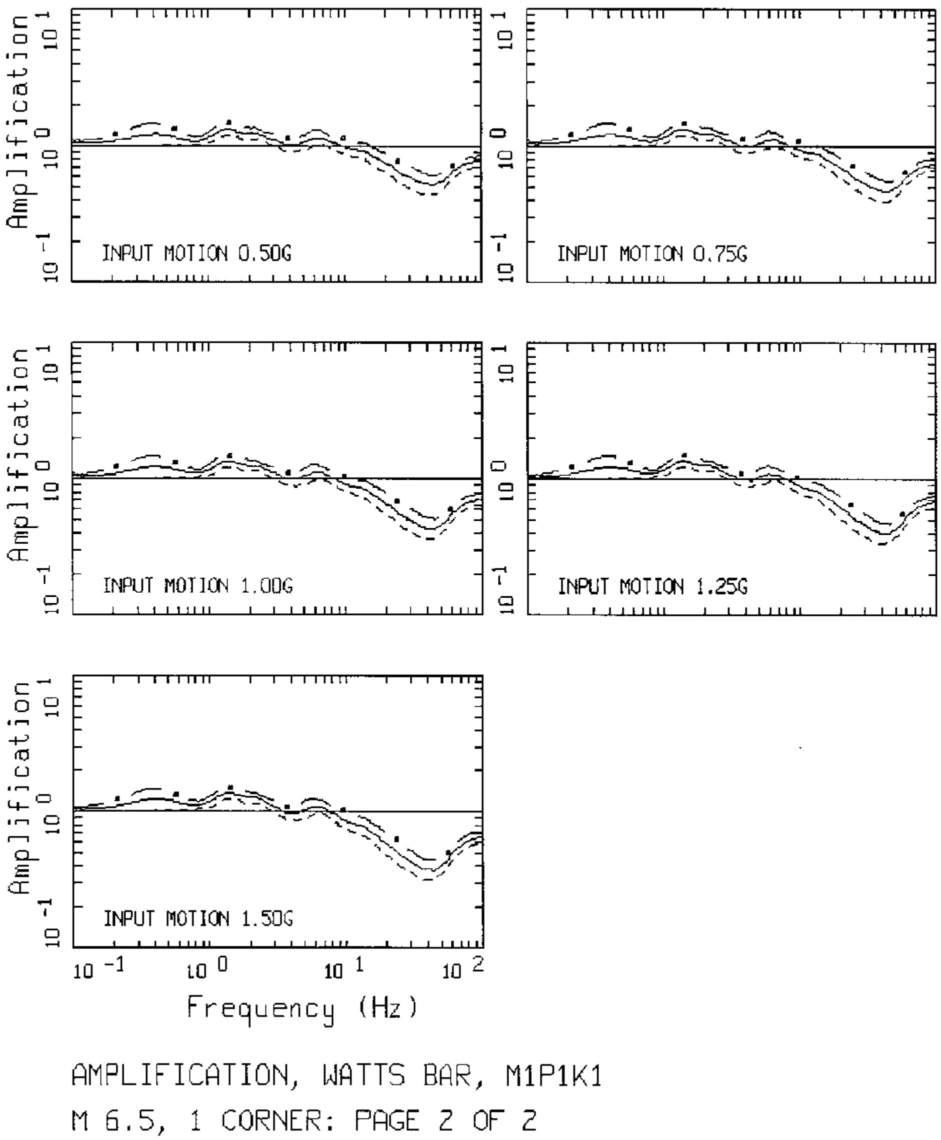


Figure 2.3.6-1.(cont.)

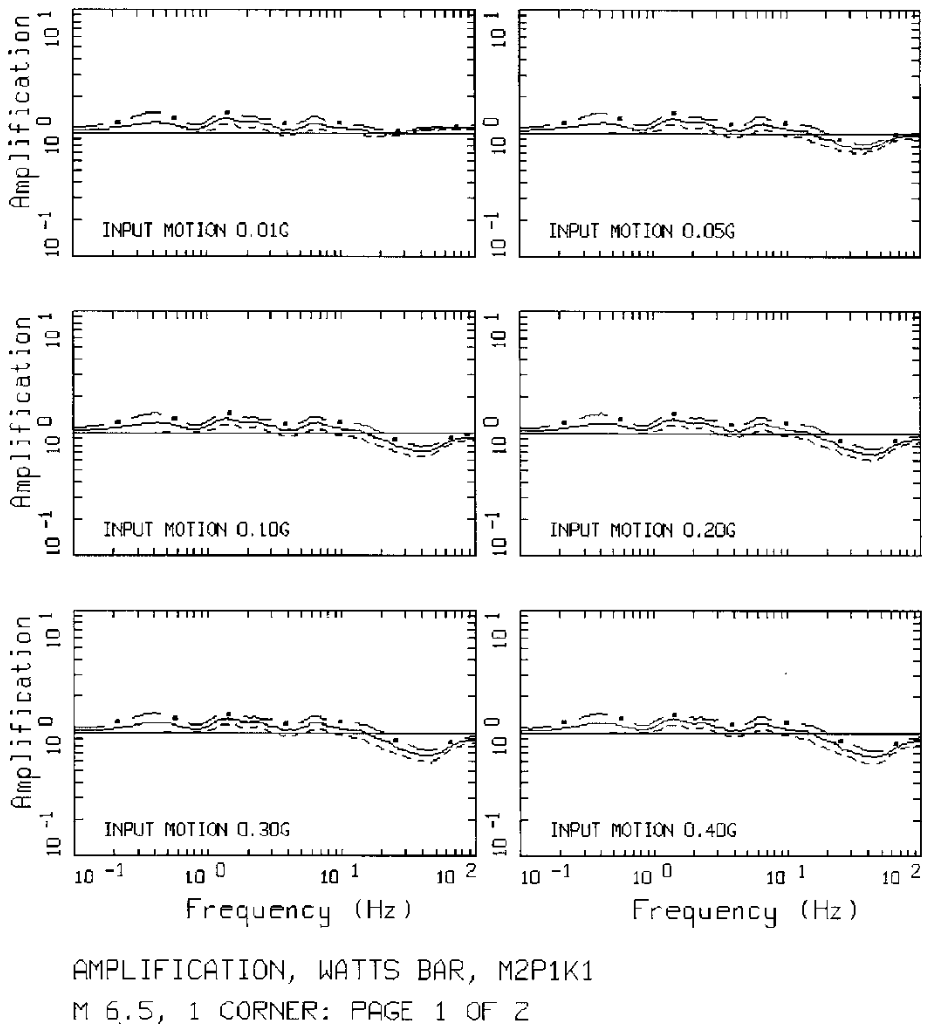


Figure 2.3.6-2. Example suite of amplification factors (5% damping pseudo absolute acceleration spectra) developed for the mean base-case profile (P1), linear site response (model M2), and base-case kappa (K1) at eleven loading levels of hard rock median peak acceleration values from 0.01g to 1.50g. **M** 6.5 and single-corner source model (EPRI, 2013a). (EPRI, 2014)

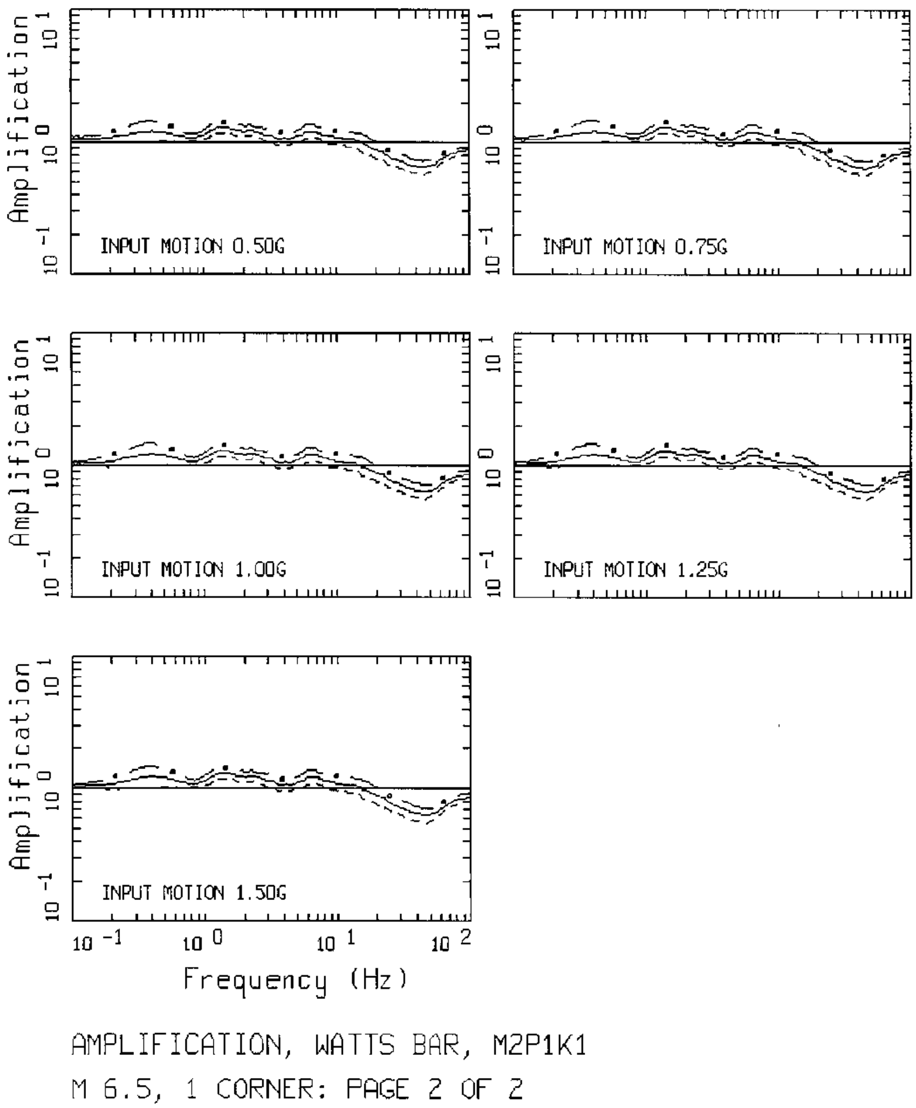


Figure 2.3.6-2.(cont.)

2.3.7 Control Point Seismic Hazard Curves

The procedure to develop probabilistic site-specific control point hazard curves used in the present analysis follows the methodology described in Section B-6.0 of the SPID (EPRI, 2013a). This procedure (referred to as Method 3) computes a site-specific control point hazard curve for a broad range of spectral accelerations given the site-specific bedrock hazard curve and site-specific estimates of soil or soft-rock response and associated uncertainties. This process is repeated for each of the seven spectral frequencies for which ground motion equations are available. The dynamic response of the materials below the control point was represented by the frequency- and amplitude-dependent amplification functions (median values and standard deviations) developed and described in the previous section. The resulting control point mean hazard curves for Watts Bar Nuclear Plant are shown in Figure 2.3.7-1 for the seven spectral frequencies for which ground motion equations are defined. Tabulated values of mean and fractile seismic hazard curves and site response amplification functions are provided in Appendix A (EPRI, 2014).

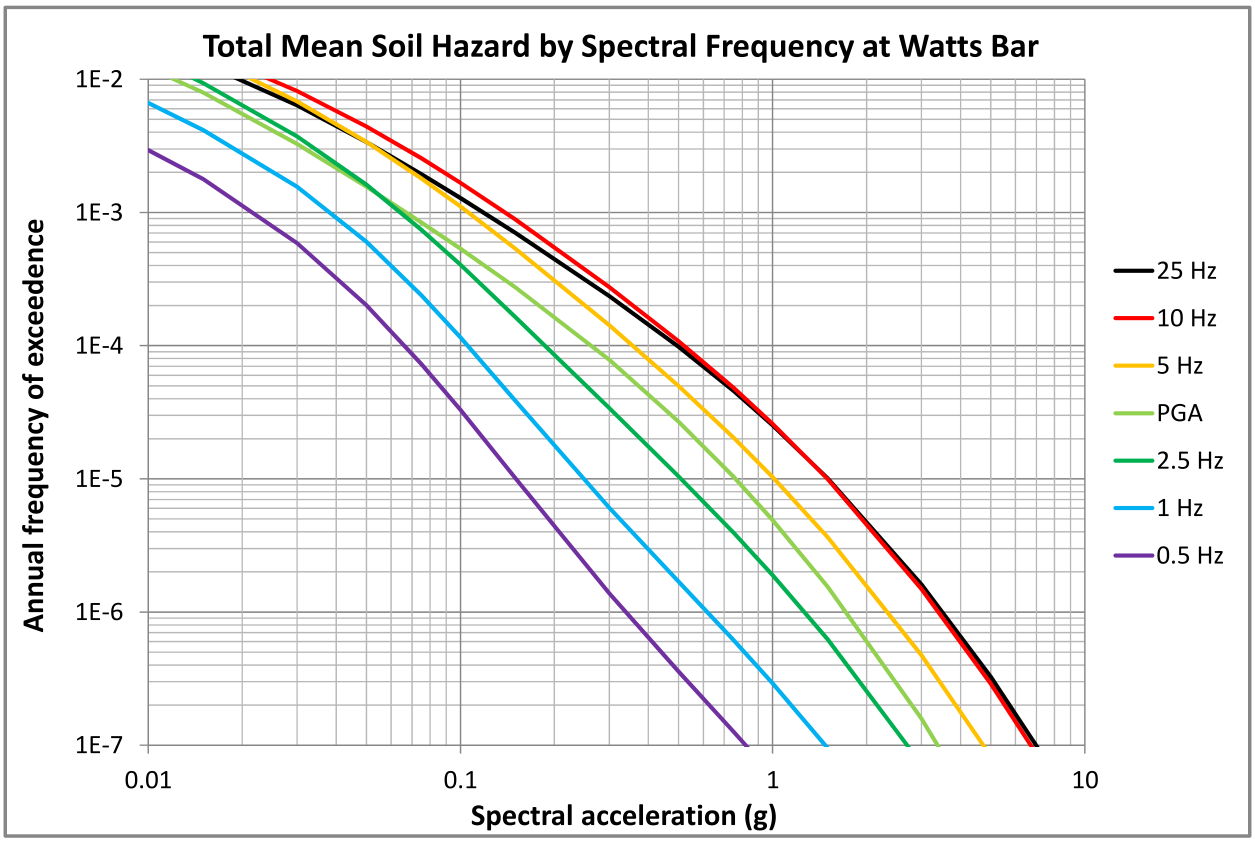


Figure 2.3.7-1. Control point mean hazard curves for spectral frequencies of 0.5, 1, 2.5, 5, 10, 25 and 100 Hz at Watts Bar Nuclear Plant (EPRI, 2014).

2.4 Control Point Response Spectrum

The control point hazard curves described above have been used to develop uniform hazard response spectra (UHRS) and the GMRS. The UHRS were obtained through linear interpolation in log-log space to estimate the spectral acceleration at each spectral frequency for the 10^{-4} and 10^{-5} per year hazard levels.

The 10^{-4} and 10^{-5} UHRS, along with a design factor (DF) are used to compute the GMRS at the control point using the criteria in Regulatory Guide 1.208 (U.S NRC, 2007). Table 2.4-1 shows the UHRS and GMRS spectral accelerations (EPRI, 2014).

Table 2.4-1. UHRS and GMRS for Watts Bar Nuclear Plant. (EPRI, 2014)

Freq. (Hz)	10^{-4} UHRS (g)	10^{-5} UHRS (g)	GMRS (g)
100	2.62E-01	7.60E-01	3.68E-01
90	2.63E-01	7.66E-01	3.71E-01
80	2.66E-01	7.79E-01	3.77E-01
70	2.73E-01	8.06E-01	3.89E-01
60	2.93E-01	8.73E-01	4.21E-01
50	3.42E-01	1.03E+00	4.95E-01
40	4.07E-01	1.23E+00	5.93E-01
35	4.39E-01	1.33E+00	6.40E-01
30	4.66E-01	1.42E+00	6.80E-01
25	4.95E-01	1.51E+00	7.23E-01
20	5.26E-01	1.58E+00	7.59E-01
15	5.39E-01	1.58E+00	7.66E-01
12.5	5.37E-01	1.56E+00	7.58E-01
10	5.19E-01	1.50E+00	7.26E-01
9	4.95E-01	1.42E+00	6.90E-01
8	4.71E-01	1.35E+00	6.55E-01
7	4.40E-01	1.26E+00	6.12E-01
6	4.04E-01	1.15E+00	5.60E-01
5	3.56E-01	1.01E+00	4.93E-01
4	2.88E-01	8.09E-01	3.95E-01
3.5	2.54E-01	7.08E-01	3.46E-01
3	2.21E-01	6.11E-01	2.99E-01
2.5	1.86E-01	5.10E-01	2.50E-01
2	1.74E-01	4.61E-01	2.28E-01
1.5	1.54E-01	3.91E-01	1.95E-01
1.25	1.34E-01	3.30E-01	1.65E-01
1	1.05E-01	2.49E-01	1.26E-01
0.9	9.54E-02	2.24E-01	1.13E-01
0.8	8.73E-02	2.04E-01	1.03E-01
0.7	8.08E-02	1.87E-01	9.51E-02
0.6	7.45E-02	1.71E-01	8.71E-02
0.5	6.61E-02	1.51E-01	7.67E-02
0.4	5.29E-02	1.21E-01	6.13E-02
0.35	4.63E-02	1.05E-01	5.37E-02
0.3	3.97E-02	9.04E-02	4.60E-02
0.25	3.31E-02	7.54E-02	3.83E-02
0.2	2.64E-02	6.03E-02	3.07E-02
0.15	1.98E-02	4.52E-02	2.30E-02
0.125	1.65E-02	3.77E-02	1.92E-02
0.1	1.32E-02	3.01E-02	1.53E-02

Figure 2.4-1 shows the control point UHRS and GMRS.

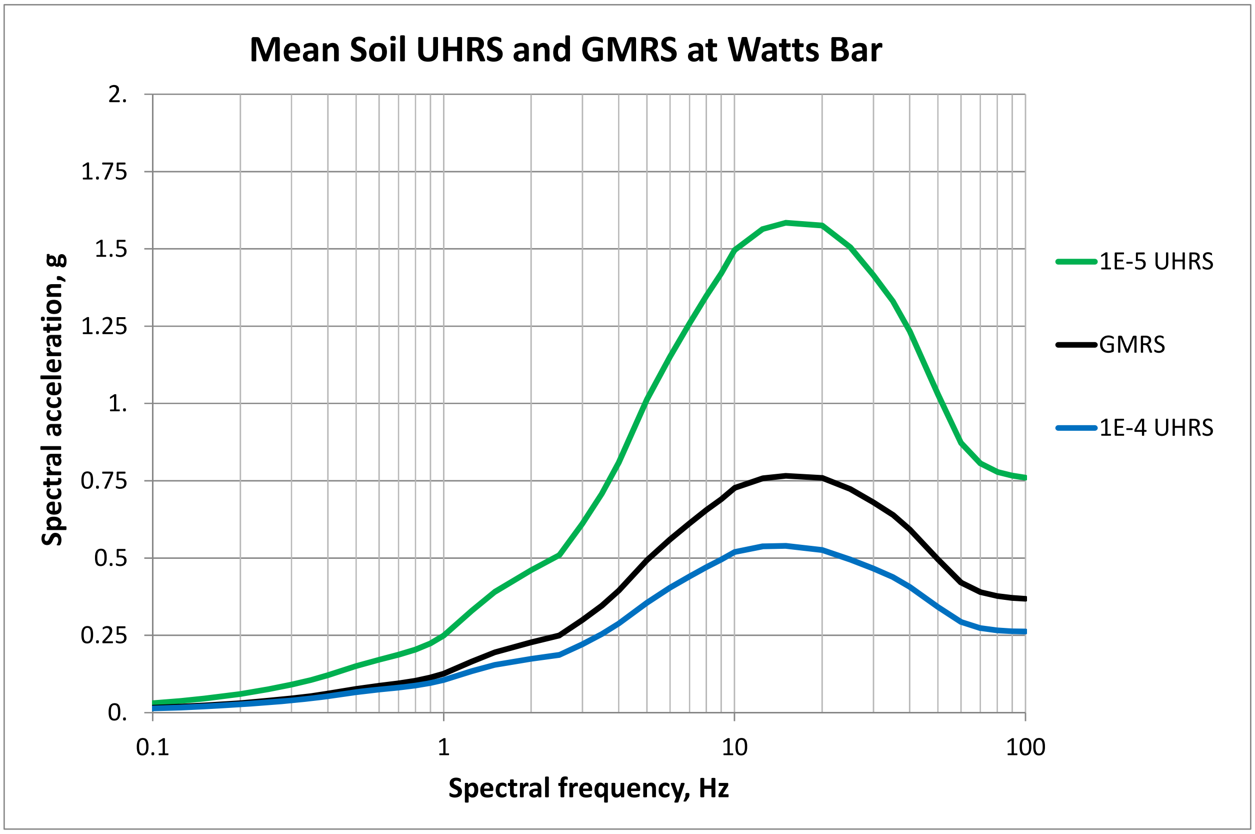


Figure 2.4-1. UHRS for 10^{-4} and 10^{-5} and GMRS at control point for Watts Bar Nuclear Plant (5%-damped response spectra). (EPRI, 2014)

3.0 Plant Design Basis Ground Motion

The design basis for Watts Bar Nuclear Plant is identified in the Updated Final Safely Evaluation Report (TVA, Amendment 11).

3.1 Safe Shutdown Earthquake Description of Spectral Shape

The SSE was developed in accordance with 10 CFR Part 100, Appendix A through an evaluation of the maximum earthquake potential for the region surrounding the site. Considering the historic seismicity of the site region, the maximum potential earthquake was determined to be intensity VIII on the Modified Mercalli Intensity Scale. Although this earthquake is listed as intensity VIII, there is considerable evidence that it should be reevaluated as an intensity MM VII (TVA, Amendment 11, Section 2.5.2.4).

The SSE is defined in terms of a PGA and a design response spectrum. Assuming an intensity of VIII occurring adjacent to the site, a PGA of 0.14 g was estimated. For additional conservatism this peak ground acceleration was increased to 0.18 g as the anchor point for the

SSE. Table 3.1-1 shows the Spectral Acceleration (SA) values as a function of frequency for the 5% damped horizontal SSE (TVA, Amendment 11, Section 2.5.2.4 and Figure 2.5-236b).

Table 3.1-1. SSE for Watts Bar Nuclear Plant (TVA, Amendment 11)

Freq (Hz)	100	33	25	10	6.67	5	2.5	2	1	0.5
SA (g)	0.18	0.18	0.22	0.36	0.46	0.46	0.46	0.46	0.22	0.11

3.2 Control Point Elevation

The SSE control point elevation is defined at the Deepest Structure Foundation Control Point, at a depth of 64 ft (Elevation 664 ft Mean Sea Level) (Table 1 of AMEC, 2013) (EPRI, 2014).

4.0 Screening Evaluation

In accordance with SPID (EPRI, 2013a) Section 3, a screening evaluation was performed as described below.

4.1 *Risk Evaluation Screening (1 to 10 Hz)*

In the 1 to 10 Hz part of the response spectrum, the GMRS exceeds the SSE. Therefore, Watts Bar Nuclear Plant screens-in for a risk evaluation.

4.2 *High Frequency Screening (> 10 Hz)*

For the range above 10 Hz, the GMRS exceeds the SSE. The high frequency exceedances can be addressed in the risk evaluation discussed in 4.1 above.

4.3 Spent Fuel Pool Evaluation Screening (1 to 10 Hz)

In the 1 to 10 Hz part of the response spectrum, the GMRS exceeds the SSE. Therefore, Watts Bar Nuclear Plant screens-in for a spent fuel pool evaluation.

5.0 Interim Actions

Based on the screening evaluation, the expedited seismic evaluation described in EPRI 3002000704 (EPRI, 2013c) will be performed as proposed in a letter to NRC (ML 13101A379) dated April 9, 2013 (NEI, 2013) and agreed to by NRC (ML 13106A331) in a letter dated May 7, 2013 (U.S. NRC, 2013a).

Consistent with NRC letter (ML 14030A046) dated February 20, 2014, (U.S. NRC, 2014a) the seismic hazard reevaluations presented herein are distinct from the current design and licensing bases of Watts Bar Nuclear Plant. Therefore, the results do not call into question the operability or functionality of SSCs and are not reportable pursuant to10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors," and10 CFR 50.73, "Licensee event report system."

The NRC letter also requests that licensees provide an interim evaluation or actions to demonstrate that the plant can cope with the reevaluated hazard while the expedited approach and risk evaluations are conducted. In response to that request, NEI letter dated March 12, 2014 (NEI, 2014), provides seismic core damage risk estimates using the updated seismic hazards for the operating nuclear plants in the Central and Eastern United States. These risk estimates continue to support the following conclusions of the NRC GI-199 Safety/Risk Assessment (U.S. NRC, 2010):

> Overall seismic core damage risk estimates are consistent with the Commission's Safety Goal Policy Statement because they are within the subsidiary objective of 10^{-4}/year for core damage frequency. The GI-199 Safety/Risk Assessment, based in part on information from the U.S. Nuclear Regulatory Commission's (NRC's) Individual Plant Examination of External Events (IPEEE) program, indicates that no concern exists regarding adequate protection and that the current seismic design of operating reactors provides a safety margin to withstand potential earthquakes exceeding the original design basis.

Watts Bar Nuclear Plant is included in the March 12, 2014 risk estimates. Using the methodology described in the NEI letter (NEI, 2014), all plants were shown to be below 10^{-4}/year; thus, the above conclusions apply.

A focus-scope Seismic Margins Assessment (SMA) was performed to support the IPEEE for Watts Bar Nuclear Plant Units 1 and 2. The results of the IPEEE for Watts Bar Nuclear Plant Units 1 and 2 were submitted to the NRC (TVA, 1998) (TVA, 2010). Results of the NRC IPEEE review are documented in the referenced Unit 1 Staff Evaluation (U.S. NRC, 2000) and Unit 2 review report (U.S. NRC, 2011).

Watts Bar Nuclear Plant Units 1 and 2 Seismic IPEEE was performed using the Seismic Margins Assessment option per the methodology of EPRI NP-6041-SLR1 (EPRI, 1991). With this method, a Seismic Margins Earthquake (SME) was postulated and the items needed for safe shutdown were then evaluated for the SME demand in two success paths (EPRI, 1991). Components and structures that were determined to have sufficient capacity to survive the SME without loss of function were screened out. Items that did not screen were subject to a more detailed evaluation, including calculation of a High Confidence of Low Probability of Failure (HCLPF) capacity PGA for that item. Watts Bar Units 1 and 2 evaluated a 0.36g HCLPF capacity and proved in general to be rugged in nature and of a sufficient capacity to provide assurance of continued functionality for the SME (TVA, 1998) (TVA, 2010).

In accordance with Near-Term Task Force Recommendation 2.3: Seismic (U.S. NRC, 2102) Watts Bar Nuclear Plant Unit 1 performed seismic walkdowns using the guidance in EPRI Report 1025286 (EPRI, 2012). The seismic walkdowns were completed and captured in the seismic walkdown report (TVA, 2012). At Watts Bar Nuclear Plant Unit 1 a total of 120 equipment items were selected from the IPEEE Safe Shutdown Equipment List (SSEL) to fulfill the requirements of the seismic walkdown guidance (TVA, 2012). The selected items were located in various environments and included many different types of equipment from multiple safety systems.

Eleven potentially adverse seismic conditions were identified and entered into the TVA Corrective Action Program (TVA, 2012). The identified potentially adverse conditions were evaluated and were found to have no operability or reportability impact on the plant. Ten out of eleven potentially adverse seismic conditions identified for Watts Bar Nuclear Plant Unit 1 have been resolved.

The seismic walkdowns (TVA, 2012) also verified in Section 7.0 that any seismic IPEEE vulnerabilities identified were adequately addressed. The seismic walkdown reports state that items walked down during the Unit 1 IPEEE program were found to be rugged and robust (TVA, 2012).

6.0 Conclusions

In accordance with the 50.54(f) request for information (U.S. NRC, 2012), a seismic hazard and screening evaluation was performed for Watts Bar Nuclear Plant. A GMRS was developed solely for purpose of screening for additional evaluations in accordance with the SPID (EPRI, 2013a). Based on the results of the screening evaluation, Watts Bar Nuclear Plant screens-in for a risk evaluation, a Spent Fuel Pool evaluation, and a High Frequency Confirmation.

References

10 CFR Part 50. Title 10, Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington DC.

10 CFR Part 100. Title 10, Code of Federal Regulations, Part 100, "Reactor Site Criteria," U.S. Nuclear Regulatory Commission, Washington DC.

10 CFR Part 50.72. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, Washington DC.

10 CFR Part 50.73. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System," U.S. Nuclear Regulatory Commission, Washington DC.

AMEC (2013). *Seismic Data Retrieval Information for EPRI Near Term Task Force Recommendation 2.1 TVA Watts Bar Nuclear Plant Spring City, Tennessee,* Letter report, AMEC Proj. 3043121029, Letter from K. Campbell to B. Enis dated October 28, 2013.

CEUS-SSC (2012). *Central and Eastern United States Seismic Source Characterization for Nuclear Facilities*, U.S. Nuclear Regulatory Commission Report, NUREG-2115; EPRI Report 1021097, 6 Volumes; DOE Report# DOE/NE-0140.

EPRI (1991). "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Revision 1, NP-6041-SLR1, August 1991.

EPRI (2012). "Seismic Walkdown Guidance: For Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic," Electric Power Research Institute, Report 1025286, June 4, 2012.

EPRI (2013a). Seismic Evaluation Guidance Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, Elec. Power Res. Inst. Rept 1025287, Feb.

EPRI (2013b). *EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project*, Elec. Power Res. Inst, Palo Alto, CA, Rept. 3002000717, June, 2 volumes.

EPRI (2013c). "Seismic Evaluation Guidance, Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," EPRI 3002000704, May 2013.

EPRI (2014). "Watts Bar Seismic Hazard and Screening Report," Electric Power Research Institute, Palo Alto, CA, dated February 7, 2014.

Hardeman, W.D., Miller, R.A., and Swingle, G.D. (1966). Geologic map of Tennessee: Tennessee Division of Geology, State Geologic Map, scale 1:250,000.

Hatcher, Robert D. Jr., Peter J. Lemiszki, and Jennifer B. Whisner (2007). Character of rigid boundaries and internal deformation of the southern Appalachian foreland fold-thrust belt Geological Society of America Special Papers, 433, p. 243-276, doi:10.1130/2007.2433(12).

Lemiszki, P.J., Kohl, M.S., and Sutton, E.F. (2008). Geologic Map and Mineral Resources Summary of the Decatur Quadrangle: Tennessee Division of Geology, Geologic Quadrangle Map 118 SE, scale 1:24,000.

NEI (2013). NEI Letter to NRC from A. Pietrangelo to D. Skeen, “Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations,” April 9, 2013.
NEI (2014). NEI Letter to NRC from A. Pietrangelo to E. Leeds, “Seismic Risk Evaluations for Plants in the Central and Eastern United States,” March 12, 2014.
Tennessee Division of Geology (1956). Bulletin 58.
Toro (1997). Appendix of: Silva, W.J., Abrahamson, N., Toro, G., and Costantino, C. (1997). “Description and validation of the stochastic ground motion model”, Report Submitted to Brookhaven National Laboratory, Associated Universities, Inc., Upton, New York 11973, Contract No. 770573.
TVA (1998). Letter from P. Pace to NRC, “Watts Bar Nuclear Plant (WBN) Unit 1 - Generic Letter 88-20, Supplements 4 and 5 - Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities (TAC No. M83693),” February 17, 1998.
TVA (2010). Letter from M. Bajestani to NRC, “Watts Barn Nuclear Plant (WBN) Unit 2 - Individual Plant Examination of External Events Design Report,” April 30, 2010.
TVA (2012). Letter from J. Shea to NRC, “Tennessee Valley Authority (TVA) - Response to NRC Request for Additional Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding the Watts Bar Nuclear Plant Seismic Walkdown Results of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident” November 27, 2012.
TVA (Amendment 11). Tennessee Valley Authority, “Watts Bar Nuclear Plant Final Safety Analysis Report,” Amendment 11.
U.S. NRC (2000). NRC Letter, R. Martin to J. Scalice, “Watts Bar Nuclear Plant, Unit 1 - Review of Individual Plant Examination of External Events (IPEEE) Submittal (TAC No. M83693),” May 19, 2000.
U.S. NRC (2007). “A performance-based approach to define the site-specific earthquake ground motion,” U.S. Nuclear Regulatory Commission Reg. Guide 1.208.
U.S. NRC (2010). “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,” GI-199, September 2, 2010.
U.S. NRC (2011). NRC Letter, S. Campbell to A. Bhatnager, “Watts Bar Nuclear Plant, Unit 2 - Review of Individual Plant Examination of External Events Design Report (TAC No. ME4482),” September 20, 2011.
U.S. NRC (2012). NRC (E Leeds and M Johnson) Letter to All Power Reactor Licensees et al., “Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident,” March 12, 2012.
U.S. NRC (2013a). NRC Letter, E. Leeds to J. Pollock, NEI “Electric Power Research Institute Final Draft Report XXXXXX, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, As an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluation,” dated May 7, 2013.
U.S. NRC (2014a). NRC Letter, E. Leeds to All Power Reactor Licensees, “Supplemental Information Related to Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Seismic Hazard Reevaluations for Recommendation 2.1 of the

Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated February 20, 2014.

Appendix A

Tabulated Data

Table A-1a. Mean and Fractile Seismic Hazard Curves for 0.5 Hz at Watts Bar Nuclear Plant (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	4.11E-02	2.07E-02	3.14E-02	4.01E-02	5.12E-02	6.00E-02
0.001	2.43E-02	1.20E-02	1.77E-02	2.29E-02	3.14E-02	3.95E-02
0.005	5.99E-03	2.04E-03	3.19E-03	5.50E-03	8.85E-03	1.16E-02
0.01	2.94E-03	5.83E-04	1.10E-03	2.53E-03	4.77E-03	6.64E-03
0.015	1.78E-03	2.35E-04	4.98E-04	1.36E-03	3.09E-03	4.70E-03
0.03	5.89E-04	3.84E-05	9.24E-05	3.28E-04	1.07E-03	2.01E-03
0.05	2.01E-04	8.60E-06	2.13E-05	8.35E-05	3.33E-04	8.12E-04
0.075	7.29E-05	2.39E-06	6.00E-06	2.53E-05	1.11E-04	3.14E-04
0.1	3.30E-05	9.11E-07	2.35E-06	1.04E-05	4.90E-05	1.42E-04
0.15	1.01E-05	2.16E-07	6.00E-07	2.80E-06	1.53E-05	4.37E-05
0.3	1.38E-06	1.29E-08	4.98E-08	2.72E-07	1.87E-06	6.45E-06
0.5	3.58E-07	1.29E-09	6.09E-09	4.70E-08	3.95E-07	1.82E-06
0.75	1.26E-07	2.60E-10	1.04E-09	1.05E-08	1.15E-07	6.26E-07
1.	5.94E-08	1.51E-10	3.42E-10	3.33E-09	4.50E-08	2.80E-07
1.5	1.93E-08	1.32E-10	1.49E-10	6.54E-10	1.07E-08	8.35E-08
3.	2.30E-09	9.11E-11	1.04E-10	1.42E-10	7.23E-10	7.34E-09
5.	3.86E-10	9.11E-11	1.01E-10	1.42E-10	1.69E-10	9.65E-10
7.5	8.09E-11	9.11E-11	1.01E-10	1.42E-10	1.42E-10	2.39E-10
10.	2.45E-11	9.11E-11	9.11E-11	1.42E-10	1.42E-10	1.49E-10

Table A-1b. Mean and Fractile Seismic Hazard Curves for 1 Hz at Watts Bar Nuclear Plant (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	8.08E-02	4.25E-02	5.75E-02	8.23E-02	9.93E-02	9.93E-02
0.001	5.41E-02	2.49E-02	3.57E-02	5.35E-02	7.13E-02	8.47E-02
0.005	1.37E-02	5.66E-03	8.23E-03	1.29E-02	1.90E-02	2.42E-02
0.01	6.65E-03	2.22E-03	3.52E-03	6.17E-03	9.65E-03	1.27E-02
0.015	4.15E-03	1.10E-03	1.87E-03	3.73E-03	6.45E-03	8.60E-03
0.03	1.56E-03	2.42E-04	4.70E-04	1.20E-03	2.64E-03	4.07E-03
0.05	6.02E-04	6.45E-05	1.32E-04	3.84E-04	1.04E-03	1.90E-03
0.075	2.39E-04	2.04E-05	4.31E-05	1.34E-04	4.07E-04	8.12E-04
0.1	1.15E-04	8.72E-06	1.90E-05	6.00E-05	1.92E-04	4.07E-04
0.15	3.86E-05	2.53E-06	5.75E-06	1.87E-05	6.36E-05	1.40E-04
0.3	6.06E-06	2.49E-07	6.64E-07	2.60E-06	9.93E-06	2.32E-05
0.5	1.69E-06	3.63E-08	1.18E-07	6.00E-07	2.72E-06	7.03E-06
0.75	6.16E-07	6.73E-09	2.64E-08	1.72E-07	9.51E-07	2.68E-06
1.	2.92E-07	1.90E-09	8.35E-09	6.45E-08	4.25E-07	1.31E-06
1.5	9.52E-08	3.52E-10	1.46E-09	1.44E-08	1.23E-07	4.37E-07
3.	1.11E-08	1.32E-10	1.57E-10	8.12E-10	1.02E-08	4.77E-08
5.	1.83E-09	9.51E-11	1.31E-10	1.67E-10	1.25E-09	6.93E-09
7.5	3.77E-10	9.11E-11	1.01E-10	1.42E-10	2.72E-10	1.29E-09
10.	1.13E-10	9.11E-11	1.01E-10	1.42E-10	1.51E-10	4.07E-10

Table A-1c. Mean and Fractile Seismic Hazard Curves for 2.5 Hz at Watts Bar Nuclear Plant (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.15E-01	8.72E-02	9.79E-02	9.93E-02	9.93E-02	9.93E-02
0.001	9.22E-02	5.91E-02	7.23E-02	9.11E-02	9.93E-02	9.93E-02
0.005	3.03E-02	1.60E-02	2.10E-02	2.92E-02	4.07E-02	4.77E-02
0.01	1.49E-02	7.34E-03	9.93E-03	1.44E-02	2.01E-02	2.42E-02
0.015	9.33E-03	4.25E-03	5.83E-03	8.98E-03	1.29E-02	1.60E-02
0.03	3.72E-03	1.23E-03	1.87E-03	3.33E-03	5.58E-03	7.45E-03
0.05	1.62E-03	4.07E-04	6.54E-04	1.31E-03	2.57E-03	3.90E-03
0.075	7.42E-04	1.53E-04	2.57E-04	5.50E-04	1.20E-03	2.01E-03
0.1	4.04E-04	7.55E-05	1.29E-04	2.88E-04	6.54E-04	1.13E-03
0.15	1.64E-04	2.76E-05	4.83E-05	1.11E-04	2.68E-04	4.70E-04
0.3	3.39E-05	4.56E-06	8.85E-06	2.25E-05	5.66E-05	1.02E-04
0.5	1.05E-05	1.05E-06	2.29E-06	6.54E-06	1.77E-05	3.37E-05
0.75	3.94E-06	2.76E-07	6.83E-07	2.25E-06	6.83E-06	1.32E-05
1.	1.89E-06	9.79E-08	2.64E-07	9.93E-07	3.28E-06	6.73E-06
1.5	6.24E-07	1.90E-08	6.00E-08	2.80E-07	1.08E-06	2.35E-06
3.	7.18E-08	8.00E-10	3.01E-09	2.10E-08	1.15E-07	3.05E-07
5.	1.13E-08	1.55E-10	3.14E-10	2.16E-09	1.57E-08	5.05E-08
7.5	2.20E-09	1.07E-10	1.42E-10	3.63E-10	2.60E-09	9.93E-09
10.	6.24E-10	9.51E-11	1.25E-10	1.62E-10	7.03E-10	2.76E-09

Table A-1d. Mean and Fractile Seismic Hazard Curves for 5 Hz at Watts Bar Nuclear Plant (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.22E-01	9.79E-02	9.93E-02	9.93E-02	9.93E-02	9.93E-02
0.001	1.05E-01	7.23E-02	8.72E-02	9.93E-02	9.93E-02	9.93E-02
0.005	4.25E-02	2.22E-02	3.09E-02	4.07E-02	5.50E-02	6.26E-02
0.01	2.28E-02	1.16E-02	1.60E-02	2.25E-02	2.96E-02	3.47E-02
0.015	1.51E-02	7.45E-03	1.02E-02	1.49E-02	2.01E-02	2.39E-02
0.03	6.82E-03	2.76E-03	4.01E-03	6.45E-03	9.65E-03	1.23E-02
0.05	3.39E-03	1.10E-03	1.67E-03	2.96E-03	5.12E-03	7.03E-03
0.075	1.80E-03	4.98E-04	7.66E-04	1.46E-03	2.84E-03	4.25E-03
0.1	1.11E-03	2.76E-04	4.31E-04	8.60E-04	1.77E-03	2.76E-03
0.15	5.35E-04	1.20E-04	1.92E-04	4.01E-04	8.72E-04	1.38E-03
0.3	1.42E-04	2.92E-05	4.90E-05	1.05E-04	2.35E-04	3.73E-04
0.5	4.99E-05	9.37E-06	1.67E-05	3.68E-05	8.35E-05	1.32E-04
0.75	2.04E-05	3.37E-06	6.36E-06	1.49E-05	3.42E-05	5.66E-05
1.	1.03E-05	1.49E-06	2.96E-06	7.23E-06	1.74E-05	2.92E-05
1.5	3.65E-06	4.07E-07	8.85E-07	2.42E-06	6.17E-06	1.10E-05
3.	4.75E-07	2.80E-08	7.23E-08	2.60E-07	8.12E-07	1.64E-06
5.	8.26E-08	2.53E-09	7.55E-09	3.52E-08	1.36E-07	3.23E-07
7.5	1.75E-08	3.73E-10	1.05E-09	5.66E-09	2.72E-08	7.34E-08
10.	5.31E-09	1.60E-10	2.92E-10	1.40E-09	7.66E-09	2.32E-08

Table A-1e. Mean and Fractile Seismic Hazard Curves for 10 Hz at Watts Bar Nuclear Plant (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.20E-01	9.51E-02	9.93E-02	9.93E-02	9.93E-02	9.93E-02
0.001	1.02E-01	6.93E-02	8.72E-02	9.93E-02	9.93E-02	9.93E-02
0.005	4.12E-02	2.29E-02	3.05E-02	4.07E-02	5.05E-02	6.45E-02
0.01	2.35E-02	1.23E-02	1.64E-02	2.29E-02	2.92E-02	3.95E-02
0.015	1.64E-02	8.00E-03	1.10E-02	1.57E-02	2.13E-02	2.88E-02
0.03	8.16E-03	3.19E-03	4.63E-03	7.45E-03	1.16E-02	1.57E-02
0.05	4.43E-03	1.40E-03	2.07E-03	3.84E-03	6.73E-03	9.65E-03
0.075	2.55E-03	6.83E-04	1.04E-03	2.07E-03	4.07E-03	6.09E-03
0.1	1.67E-03	4.01E-04	6.17E-04	1.29E-03	2.72E-03	4.19E-03
0.15	8.85E-04	1.92E-04	3.01E-04	6.54E-04	1.44E-03	2.32E-03
0.3	2.75E-04	5.42E-05	9.11E-05	1.98E-04	4.56E-04	7.45E-04
0.5	1.08E-04	2.01E-05	3.52E-05	7.89E-05	1.79E-04	2.92E-04
0.75	4.81E-05	8.12E-06	1.51E-05	3.47E-05	8.00E-05	1.32E-04
1.	2.59E-05	3.95E-06	7.66E-06	1.84E-05	4.31E-05	7.23E-05
1.5	9.94E-06	1.21E-06	2.60E-06	6.83E-06	1.69E-05	2.92E-05
3.	1.49E-06	1.02E-07	2.60E-07	8.85E-07	2.60E-06	4.90E-06
5.	2.90E-07	1.05E-08	3.14E-08	1.38E-07	5.05E-07	1.08E-06
7.5	6.72E-08	1.34E-09	4.50E-09	2.53E-08	1.13E-07	2.76E-07
10.	2.18E-08	3.33E-10	1.05E-09	6.64E-09	3.47E-08	9.37E-08

Table A-1f. Mean and Fractile Seismic Hazard Curves for 25 Hz at Watts Bar Nuclear Plant (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.09E-01	6.45E-02	9.51E-02	9.93E-02	9.93E-02	9.93E-02
0.001	8.56E-02	4.43E-02	7.13E-02	8.60E-02	9.93E-02	9.93E-02
0.005	3.17E-02	1.62E-02	2.25E-02	3.01E-02	3.79E-02	6.17E-02
0.01	1.87E-02	8.60E-03	1.21E-02	1.74E-02	2.35E-02	3.90E-02
0.015	1.32E-02	5.42E-03	7.77E-03	1.21E-02	1.74E-02	2.76E-02
0.03	6.39E-03	1.92E-03	2.92E-03	5.50E-03	9.65E-03	1.44E-02
0.05	3.38E-03	7.66E-04	1.23E-03	2.64E-03	5.42E-03	8.60E-03
0.075	1.94E-03	3.73E-04	6.17E-04	1.38E-03	3.19E-03	5.42E-03
0.1	1.28E-03	2.29E-04	3.84E-04	8.85E-04	2.10E-03	3.73E-03
0.15	7.03E-04	1.20E-04	2.01E-04	4.77E-04	1.15E-03	2.07E-03
0.3	2.36E-04	3.84E-05	6.83E-05	1.60E-04	3.90E-04	6.83E-04
0.5	9.81E-05	1.49E-05	2.76E-05	6.73E-05	1.64E-04	2.84E-04
0.75	4.57E-05	6.17E-06	1.21E-05	3.09E-05	7.77E-05	1.34E-04
1.	2.53E-05	3.01E-06	6.36E-06	1.69E-05	4.31E-05	7.66E-05
1.5	1.01E-05	9.79E-07	2.22E-06	6.36E-06	1.74E-05	3.23E-05
3.	1.61E-06	8.98E-08	2.42E-07	8.47E-07	2.80E-06	5.75E-06
5.	3.26E-07	9.93E-09	3.19E-08	1.34E-07	5.58E-07	1.25E-06
7.5	7.80E-08	1.38E-09	4.90E-09	2.46E-08	1.29E-07	3.23E-07
10.	2.58E-08	3.63E-10	1.20E-09	6.45E-09	4.01E-08	1.10E-07

Table A-1g. Mean and Fractile Seismic Hazard Curves for PGA at Watts Bar Nuclear Plant (EPRI, 2014)

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.00E-01	4.83E-02	8.35E-02	9.93E-02	9.93E-02	9.93E-02
0.001	7.35E-02	3.23E-02	5.83E-02	7.23E-02	9.24E-02	9.93E-02
0.005	2.28E-02	1.07E-02	1.55E-02	2.16E-02	2.80E-02	4.63E-02
0.01	1.22E-02	4.77E-03	7.13E-03	1.11E-02	1.62E-02	2.80E-02
0.015	7.94E-03	2.60E-03	3.95E-03	6.83E-03	1.13E-02	1.92E-02
0.03	3.27E-03	7.45E-04	1.15E-03	2.46E-03	5.20E-03	9.37E-03
0.05	1.56E-03	2.84E-04	4.43E-04	1.04E-03	2.53E-03	4.83E-03
0.075	8.42E-04	1.40E-04	2.22E-04	5.27E-04	1.34E-03	2.68E-03
0.1	5.35E-04	8.60E-05	1.42E-04	3.33E-04	8.47E-04	1.69E-03
0.15	2.74E-04	4.25E-05	7.34E-05	1.72E-04	4.37E-04	8.60E-04
0.3	7.83E-05	1.01E-05	1.95E-05	4.90E-05	1.27E-04	2.42E-04
0.5	2.70E-05	2.57E-06	5.75E-06	1.64E-05	4.43E-05	8.47E-05
0.75	1.03E-05	6.83E-07	1.74E-06	5.91E-06	1.74E-05	3.47E-05
1.	4.91E-06	2.25E-07	6.54E-07	2.57E-06	8.35E-06	1.72E-05
1.5	1.55E-06	3.63E-08	1.32E-07	6.73E-07	2.64E-06	5.91E-06
3.	1.58E-07	8.12E-10	4.56E-09	4.13E-08	2.42E-07	6.83E-07
5.	2.18E-08	1.42E-10	3.09E-10	3.23E-09	2.84E-08	1.02E-07
7.5	3.67E-09	1.01E-10	1.42E-10	4.13E-10	3.95E-09	1.74E-08
10.	9.16E-10	9.11E-11	1.13E-10	1.62E-10	9.11E-10	4.50E-09

Table A-2. Amplification Functions for Watts Bar Nuclear Plant (EPRI, 2014)

PGA	Median AF	Sigma ln(AF)	25 Hz	Median AF	Sigma ln(AF)	10 Hz	Median AF	Sigma ln(AF)	5 Hz	Median AF	Sigma ln(AF)
1.00E-02	1.07E+00	4.34E-02	1.30E-02	9.60E-01	5.71E-02	1.90E-02	1.04E+00	9.54E-02	2.09E-02	1.13E+00	1.14E-01
4.95E-02	9.33E-01	6.14E-02	1.02E-01	7.78E-01	1.10E-01	9.99E-02	1.02E+00	1.11E-01	8.24E-02	1.12E+00	1.17E-01
9.64E-02	8.84E-01	6.79E-02	2.13E-01	7.47E-01	1.21E-01	1.85E-01	1.01E+00	1.13E-01	1.44E-01	1.12E+00	1.16E-01
1.94E-01	8.42E-01	7.36E-02	4.43E-01	7.21E-01	1.28E-01	3.56E-01	9.94E-01	1.14E-01	2.65E-01	1.11E+00	1.16E-01
2.92E-01	8.19E-01	7.68E-02	6.76E-01	7.06E-01	1.32E-01	5.23E-01	9.85E-01	1.15E-01	3.84E-01	1.11E+00	1.16E-01
3.91E-01	8.04E-01	7.91E-02	9.09E-01	6.93E-01	1.35E-01	6.90E-01	9.77E-01	1.16E-01	5.02E-01	1.10E+00	1.15E-01
4.93E-01	7.91E-01	8.11E-02	1.15E+00	6.82E-01	1.37E-01	8.61E-01	9.70E-01	1.17E-01	6.22E-01	1.10E+00	1.15E-01
7.41E-01	7.68E-01	8.48E-02	1.73E+00	6.59E-01	1.42E-01	1.27E+00	9.55E-01	1.20E-01	9.13E-01	1.09E+00	1.16E-01
1.01E+00	7.50E-01	8.71E-02	2.36E+00	6.41E-01	1.46E-01	1.72E+00	9.41E-01	1.22E-01	1.22E+00	1.08E+00	1.16E-01
1.28E+00	7.36E-01	8.83E-02	3.01E+00	6.25E-01	1.48E-01	2.17E+00	9.29E-01	1.24E-01	1.54E+00	1.08E+00	1.17E-01
1.55E+00	7.25E-01	8.90E-02	3.63E+00	6.11E-01	1.49E-01	2.61E+00	9.18E-01	1.26E-01	1.85E+00	1.07E+00	1.18E-01

2.5 Hz	Median AF	Sigma ln(AF)	1 Hz	Median AF	Sigma ln(AF)	0.5 Hz	Median AF	Sigma ln(AF)
2.18E-02	1.08E+00	1.03E-01	1.27E-02	1.25E+00	1.09E-01	8.25E-03	1.20E+00	1.42E-01
7.05E-02	1.07E+00	1.02E-01	3.43E-02	1.24E+00	1.05E-01	1.96E-02	1.20E+00	1.37E-01
1.18E-01	1.07E+00	1.02E-01	5.51E-02	1.24E+00	1.03E-01	3.02E-02	1.20E+00	1.35E-01
2.12E-01	1.07E+00	1.01E-01	9.63E-02	1.24E+00	1.02E-01	5.11E-02	1.20E+00	1.34E-01
3.04E-01	1.06E+00	9.97E-02	1.36E-01	1.24E+00	1.02E-01	7.10E-02	1.20E+00	1.34E-01
3.94E-01	1.06E+00	9.90E-02	1.75E-01	1.24E+00	1.01E-01	9.06E-02	1.20E+00	1.33E-01
4.86E-01	1.06E+00	9.85E-02	2.14E-01	1.24E+00	1.01E-01	1.10E-01	1.20E+00	1.33E-01
7.09E-01	1.06E+00	9.75E-02	3.10E-01	1.25E+00	1.01E-01	1.58E-01	1.20E+00	1.33E-01
9.47E-01	1.06E+00	9.65E-02	4.12E-01	1.25E+00	1.01E-01	2.09E-01	1.20E+00	1.33E-01
1.19E+00	1.06E+00	9.59E-02	5.18E-01	1.25E+00	1.01E-01	2.62E-01	1.20E+00	1.33E-01
1.43E+00	1.06E+00	9.59E-02	6.19E-01	1.25E+00	1.01E-01	3.12E-01	1.20E+00	1.33E-01

Tables A-3a and A-3b are tabular versions of the typical amplification factors provided in Figures 2.3.6-1 and 2.3.6-2. Values are provided for two input motion levels at approximately 10^{-4} and 10^{-5} mean annual frequency of exceedance. These factors are unverified and are provided for information only. The figures should be considered the governing information.

Table A-3a. Median AFs and sigmas for Model 1, Profile 1, for 2 PGA levels. (EPRI, 2014)

For Information Only

M1P1K1	Rock PGA=0.292			M1P1K1	PGA=1.01		
Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)	Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)
100.0	0.243	0.832	0.084	100.0	0.710	0.707	0.099
87.1	0.246	0.817	0.085	87.1	0.716	0.689	0.101
75.9	0.250	0.791	0.088	75.9	0.725	0.658	0.104
66.1	0.258	0.741	0.095	66.1	0.742	0.602	0.110
57.5	0.273	0.664	0.108	57.5	0.775	0.520	0.123
50.1	0.302	0.607	0.133	50.1	0.836	0.459	0.147
43.7	0.340	0.577	0.155	43.7	0.922	0.428	0.171
38.0	0.379	0.588	0.165	38.0	1.015	0.436	0.186
33.1	0.417	0.615	0.173	33.1	1.122	0.464	0.199
28.8	0.444	0.659	0.171	28.8	1.208	0.508	0.200
25.1	0.466	0.690	0.156	25.1	1.282	0.544	0.192
21.9	0.494	0.773	0.160	21.9	1.360	0.616	0.187
19.1	0.516	0.822	0.153	19.1	1.443	0.674	0.182
16.6	0.527	0.880	0.149	16.6	1.496	0.739	0.165
14.5	0.543	0.951	0.136	14.5	1.547	0.811	0.155
12.6	0.541	0.979	0.119	12.6	1.573	0.858	0.134
11.0	0.538	1.003	0.117	11.0	1.563	0.884	0.126
9.5	0.531	1.040	0.121	9.5	1.566	0.937	0.131
8.3	0.510	1.085	0.108	8.3	1.540	1.009	0.116
7.2	0.497	1.132	0.100	7.2	1.507	1.064	0.099
6.3	0.480	1.168	0.127	6.3	1.461	1.107	0.120
5.5	0.440	1.124	0.128	5.5	1.357	1.086	0.138
4.8	0.406	1.062	0.098	4.8	1.250	1.030	0.115
4.2	0.385	1.041	0.111	4.2	1.168	0.999	0.111
3.6	0.371	1.033	0.107	3.6	1.130	1.000	0.105
3.2	0.377	1.118	0.108	3.2	1.146	1.083	0.107
2.8	0.365	1.143	0.094	2.8	1.125	1.127	0.091
2.4	0.357	1.212	0.073	2.4	1.111	1.213	0.071
2.1	0.325	1.215	0.098	2.1	1.024	1.236	0.094
1.8	0.293	1.228	0.112	1.8	0.925	1.255	0.109
1.6	0.263	1.275	0.089	1.6	0.829	1.304	0.088
1.4	0.233	1.313	0.108	1.4	0.730	1.341	0.106
1.2	0.197	1.261	0.098	1.2	0.612	1.285	0.097
1.0	0.165	1.174	0.101	1.0	0.509	1.193	0.101
0.91	0.141	1.106	0.062	0.91	0.431	1.120	0.063
0.79	0.125	1.086	0.065	0.79	0.379	1.097	0.065
0.69	0.113	1.104	0.091	0.69	0.339	1.112	0.090
0.60	0.101	1.142	0.120	0.60	0.302	1.148	0.119
0.52	0.089	1.181	0.149	0.52	0.264	1.186	0.147
0.46	0.076	1.206	0.170	0.46	0.223	1.210	0.170
0.10	0.003	1.060	0.049	0.10	0.008	1.052	0.048

Table A-3b. Median AFs and sigmas for Model 2, Profile 1, for 2 PGA levels. (EPRI, 2014)
For Information Only

M2P1K1	PGA=0.292			M2P1K1	PGA=1.01		
Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)	Freq. (Hz)	Soil_SA	med. AF	sigma ln(AF)
100.0	0.260	0.890	0.066	100.0	0.866	0.861	0.070
87.1	0.263	0.876	0.067	87.1	0.879	0.845	0.071
75.9	0.269	0.851	0.068	75.9	0.901	0.817	0.073
66.1	0.279	0.804	0.071	66.1	0.944	0.765	0.077
57.5	0.300	0.730	0.079	57.5	1.029	0.690	0.087
50.1	0.339	0.681	0.099	50.1	1.182	0.649	0.109
43.7	0.387	0.658	0.116	43.7	1.367	0.635	0.126
38.0	0.433	0.672	0.117	38.0	1.528	0.656	0.125
33.1	0.475	0.700	0.125	33.1	1.669	0.689	0.132
28.8	0.502	0.744	0.121	28.8	1.750	0.735	0.125
25.1	0.523	0.775	0.125	25.1	1.810	0.767	0.128
21.9	0.552	0.863	0.145	21.9	1.891	0.857	0.148
19.1	0.567	0.903	0.141	19.1	1.925	0.899	0.143
16.6	0.575	0.959	0.153	16.6	1.936	0.956	0.155
14.5	0.586	1.028	0.136	14.5	1.958	1.026	0.138
12.6	0.580	1.050	0.121	12.6	1.922	1.048	0.122
11.0	0.576	1.072	0.117	11.0	1.893	1.070	0.117
9.5	0.560	1.097	0.118	9.5	1.830	1.095	0.119
8.3	0.531	1.130	0.110	8.3	1.722	1.128	0.110
7.2	0.515	1.174	0.100	7.2	1.661	1.172	0.100
6.3	0.495	1.204	0.123	6.3	1.587	1.203	0.123
5.5	0.450	1.150	0.116	5.5	1.437	1.149	0.116
4.8	0.414	1.084	0.090	4.8	1.316	1.084	0.090
4.2	0.394	1.065	0.111	4.2	1.244	1.064	0.111
3.6	0.378	1.052	0.110	3.6	1.188	1.051	0.110
3.2	0.383	1.136	0.109	3.2	1.201	1.135	0.109
2.8	0.369	1.156	0.100	2.8	1.153	1.155	0.100
2.4	0.358	1.218	0.075	2.4	1.114	1.216	0.075
2.1	0.325	1.215	0.098	2.1	1.006	1.214	0.098
1.8	0.292	1.225	0.112	1.8	0.902	1.223	0.111
1.6	0.262	1.271	0.089	1.6	0.807	1.269	0.089
1.4	0.232	1.309	0.108	1.4	0.711	1.307	0.107
1.2	0.196	1.257	0.097	1.2	0.598	1.255	0.096
1.0	0.164	1.171	0.100	1.0	0.499	1.170	0.098
0.91	0.141	1.104	0.060	0.91	0.425	1.104	0.060
0.79	0.125	1.085	0.065	0.79	0.375	1.085	0.064
0.69	0.113	1.103	0.091	0.69	0.336	1.104	0.090
0.60	0.101	1.142	0.120	0.60	0.300	1.141	0.118
0.52	0.089	1.181	0.148	0.52	0.262	1.180	0.147
0.46	0.076	1.206	0.170	0.46	0.222	1.206	0.169
0.10	0.003	1.060	0.049	0.10	0.008	1.050	0.047